VERMONT YANKEE NUCLEAR POWER CORPORATION

Ferry Road, Brattleboro, VT 05301-7002

REPLY TO
ENGINEERING OFFICE
580 MAIN STREET
BOLTON, MA 01740
(508) 779-6711

February 28, 1990

BVY 90-018

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

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

Subject: Vermont Yankee 1989 Annual Operating Report

Dear Sir:

Enclosed please find one copy of the Vermont Yankee Nuclear Power Corporation Annual Operating Report submitted in accordance with 10CFR50.59(b). This report describes the facility changes, tests, and experiments conducted without prior NRC approval during the year 1989.

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

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Leonard A. Tremblay, Jr.

L. A. Tremblay, Jr.
Senior Licensing Engineer

LAT/sv

cc: USNRC, Region I Administrator
USNRC, Project Manager - VYNPS
USNRC, Resident Inspector - VYNPS

IE47
1/1

OPERATIONS SUMMARY

Between January 1 and December 31 of 1989, Vermont Yankee completed a number of changes. The following report describes those changes which constituted a change in the facility as described in the Final Safety Analysis Report (FSAR). The report includes eleven (11) Engineering Design Change Requests (EDCR), seven (7) Plant Design Change Requests (PDCR), three (3) Plant Alteration Requests (PAR), six (6) Temporary Modifications (TM), eight (8) Temporary Mechanical Bypass Requests (MBR), four (4) Temporary Lifted Lead and Jumper Requests (LL/JR), one (1) Test Procedure, and two (2) Special Test Procedures (STP). There were no Safety and Relief Valve Challenges and/or Failures during the 1989 calendar year.

A. Changes in Facility Design

1. During 1989 there were no changes made which required authorization from the Commission.

2. The following changes did not require Commission approval, they were reviewed by the Plant Operations Review Committee (PORC), and approved by the Plant Manager and the Manager of Operations. It was determined that these changes did not involve unreviewed safety questions as defined in 10 CFR 50.59(a)(2).

   (a) EDCR 88-403, "Feedwater Heater Modification and Repair" was completed April 19, 1989.

   GENERAL SUMMARY:

   Feedwater Heater No. 3 at Vermont Yankee is a horizontal U-tube heat exchanger designed for handling condensing steam from the turbine, as well as drain cooling. Normal operating conditions on the steam side are 70.8 psia at 303.7°f. There are two identical units, one in each feedwater train.

   In June 1988, a steam leak developed in the heater shell near the 24" steam inlet nozzle. Subsequent inspection revealed shell thinning due to steam erosion. A similar condition was identified in the steam inlet nozzle. The second heat exchanger showed similar wall thinning, but was not leaking.

   A temporary repair was accomplished by sleeving the nozzle and weld overlay repairing the shell (repair was performed on both heat exchangers).

   Further analyses were conducted, with the conclusion being that the weld overlay repair would be considered a permanent repair, but the nozzles should be replaced.

   A nozzle modification restored the steam inlet nozzle to its required wall thickness, and provided a chrome-moly steel flow path for improved erosion resistance. This design change achieved the following:

1. Steam inlet velocity to the heat exchanger was unchanged since a 24" steam inlet was maintained.

2. All welds are separated by at least $2.5\sqrt{rt}$, minimizing additive residual stress effects.

3. The entire steam flow path into the heat exchanger is chrome-moly steel.

SAFETY EVALUATION SUMMARY:

The steam inlet nozzles are described in the FSAR to the extent that Figure 11.2-1 shows the nozzles to be 24" in diameter. This design change generated a change to the FSAR which revised the diameter of the nozzles to 30". The text of FSAR Chapter 11 states that the vessels will be designed and fabricated to the requirements of ASME Section VIII. This design change was completed in accordance with the requirements of the original construction code.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(b) EDCR 88-402, "MOV Control Wire Modification" was completed March 29, 1989.

GENERAL SUMMARY:

Routine testing revealed that motor-operated valves can be damaged due to thrusting into the backseat under torque. The backseat torque, during the opening stroke of the valve, can exceed the manufacturer's recommended value, and over stress the valve stem. The excessive torque is caused by motor-operator inertia at the end of the open cycle. This is also referred to as coastdown. The motor operator continues to run until the "open" torque switch operates to de-energize the motor when the valve disk hits the backseat. The motor, however, has enough inertia to provide excessive torque. In order to prevent the ove torque condition from occurring, a 95 percent open limit switch contact was connected in series with the torque switch for each valve. This limit switch contact will operate to de-energize the motor operator on the open stroke when the valve is approximately 95 percent open. Motor coastdown will provide the additional valve travel to fully open the valve without excessive backseating torque. Additionally, to ensure that the valve opens on demand, an open limit switch was connected in parallel with the open torque switch to bypass the torque switch during the beginning of the open stroke. This will permit maximum torque for unseating the valve during the beginning of the opening cycle.

SAFETY EVALUATION SUMMARY:

For valves which are normally closed, and must open upon an accident signal, this change will help ensure performance of their safety function. The change will also help protect the valves from internal damage and enable the valves to reposition. The addition of the open limit switch in parallel with the open torque switch will ensure that maximum torque can be applied to unseat the valves at the beginning of the open stroke. This change will also modify the present open-circuit wiring which allows the limit switch to bypass the torque switch during the entire stroke, thus preventing the torque switch from protecting the valve for mid-travel obstructions. The addition of the 95 percent open limit switch in series with the open torque switch protects the valves from internal damage by ensuring that excessive torque is not applied to the valve backseat at the end of the open stroke. The 95 percent open limit switch contact will de-energize the motor operator before the valve is fully open, allowing motor coastdown to supply the additional valve travel to fully open the valve. The 95 percent value is approximate and will be set in the field to allow full valve opening after motor coastdown. However, even if 100 percent valve opening were not achieved, there is negligible flow difference between 95 to 100 percent valve open.

This alteration did not present significant hazards not described or implicit in the Safety Analysis Report and there is reasonable assurance that the health and safety of the public was not endangered.

(c) EDCR 88-401, ECN 2, "HPCI Pump Impeller Replacement", was completed April 12, 1989.

GENERAL SUMMARY:

Over the past several years, Vermont Yankee has implemented a number of design changes which have significantly improved the seismic capabilities of safety-related piping systems (e.g., Torus Attached Piping Modifications, Seismic Reanalysis Program). Upon completion of these programs, high vibration readings (occasionally in the "Alert" range) were recorded on the HPCI pump train during normal monthly surveillance activities. In 1984, a program was instituted to evaluate the nozzles on rotating equipment in the plant which are safety related. This ensured that the pipe support changes made during the various seismic upgrade programs did not adversely affect allowable nozzle loadings and ruled out the pipe support changes as a cause for high vibration readings.

The loads on the HPCI turbine inlet nozzle were above the manufacturer's recommended allowables. It was also determined during this time that the HPCI turbine was misaligned with the high pressure pump. It seemed logical that the high nozzle loads created the misalignment which caused the high vibrations in the high pressure pump.

The HPCI turbine inlet line was reanalyzed and resupported to reduce nozzle loads to acceptable levels (VYC-369), and the turbine was realigned with the pump train. However, the high vibrations in the high pressure pump persisted.

Based upon recommendations from Yankee Nuclear Services Division, the pump vendor, and other special consultants under contract to Vermont Yankee, Vermont Yankee decided to eliminate the forcing function of the vibration by replacing the existing four-vane impeller of the low pressure (booster) pump with a new five-vane impeller.

SAFETY EVALUATION SUMMARY:

This design change had no effect on the operation characteristics of the HPCI System, nor did it change its ability to meet its safety objectives as defined in the Final Safety Analysis Report. The additional vane in the impeller, along with the added 1/2" to the diameter, provides the same flow requirements stated in the FSAR with a reduced net positive suction head required. Furthermore, the elimination of the forcing function creating the noticeable vibration at running speed will enable the system to perform its design function more efficiently. Additionally, the elimination of the impeller wear rings and the betterment of all the materials used in the rotating element means that there is less probability of a malfunction.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(d) EDCR 87-413, "APRM/LPRM Power Supply Modification", was completed 03/30/89.

GENERAL SUMMARY:

During development of the Vermont Yankee Environmental Qualification (EQ) Program, the need to verify a successful plant scram following a loss-of-coolant accident (post-LOCA) was identified. The position taken in the EQ Program was consistent with Regulatory Guide 1.97, which also identified the need to verify a successful plant scram post-LOCA. To meet this requirement, the Local Power Range Monitors (LPRMs) were qualified for ten minutes post-LOCA operation by EDCR 84-427. However, it was pointed out that the Average Power Range Monitors (APRMs), to which the LPRMs supply signals, are powered by the Reactor Protection System Motor Generator (RPS MG) set. Following a loss of off-site power, the RPS MG set trips, which removes power from the APRMs and, with them, the LPRMs. This action negates the Post-Accident Monitoring (PAM) function being performed by the LPRMs.

Verification of reactor shutdown is the operator's first priority following a plant trip. The controls which must be reset to restart the MG sets are located in the cable vault, which would require an operator to leave the Control Room at a time when he is needed in the Control Room to assess what has happened. To correct this problem, an alternate source of power was made available to the APRMs to power them for PAM usage. This design change detailed the modifications necessary to provide the APRMs with an alternate power source, thereby ensuring that the LPRMs accomplish their intended PAM function.

SAFETY EVALUATION SUMMARY:

The APRMs and LPRMs are safety class electrical and perform two safety functions:

1. Scram the reactor on high power.

2. Provide post-accident shutdown verification.

However, the normal power sources for the APRMs are not available post-LOCA to accomplish Safety Function 2. Consequently, this design change installed power transfer contactors in the APRM power feed circuitry. These contactors will automatically repower the APRMs from an alternate power source if the normal power source becomes inoperable.

The normal power source for the two APRM buses will continue to be RPS A or B. Powering the APRMs from the RPS MG sets assures that the APRMs perform their intended RPS function of creating a scram on high reactor power or on loss of RPS power. Once the RPS scram has been generated, the APRMs have accomplished their intended function and are not longer needed by RPS.

Following a loss of off-site power, the RPS MG sets trip off-line and the RPS buses de-energize. Once de-energized, the contactors change state to power the APRM from the alternate source. This action allows the APRMs and associated LPRMs to accomplish their intended post-accident monitoring function.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(e) EDCR 87-412, "Diesel Fire Pump Injection Path Modifications", was completed February 3, 1989.

GENERAL SUMMARY:

One of the improvements suggested in the Vermont Yankee containment Safety Study was to use the diesel fire pump to provide coolant (spray or injection) to the containment or torus during station blackout conditions. Using the existing piping connections, only valve repositioning is required to provide the proper flow path. Existing Emergency Operating Procedure OE-3102, Appendix B, "Alternate Injection Using Fire System to RHR", provides a step-by-step instruction for using the fire water pumps to inject river water to the vessel. However, most of the required valves have AC motor operators which could not be electrically operated during a station blackout. Those valves which are located outside of the Reactor Building could be operated manually, but is is desirable to operate those valves located in the Reactor Building remotely because of the postulated harsh environment which develops as a result of the station blackout. Furthermore, some of these valves would be difficult to operate manually because of their location.

This design change provided a means to operate the subject valves using the John Deere Diesel Generator (JDDG). In addition, the ability to use the JDDG to recharge the station batteries was also provided.

SAFETY EVALUATION SUMMARY:

This design change consisted mainly of engineering analysis, and made use of existing plant capabilities. The only impact on safety class equipment was the electrical cable connection made to the spare Compartment 6D of Bus 9. Under normal plant operating conditions, there is no circuit breaker in Compartment 6D; therefore, there is no way for an inadvertent electrical connection to be made between AC-DP-D1A and Bus 9. The empty breaker Compartment 6D guarantees separation between the safety class Bus 9 and the non-safety distribution panel, AC-DP-D1A. Further separation was provided by leaving the feeder breaker in AC-DP-D1A open.

This design change increased plant safety by providing the means to use the diesel fire pump to inject coolant spray during station blackout conditions. The use of the JDDG to operate loads on MCC 9B and MCC 8B is intended only as a last resort during station blackout conditions and is not intended to be used during design basis conditions. The separation between the safety class and non-safety electrical systems, provided by the empty circuit breaker compartment in Bus 9, will prevent this design change from adversely impacting the safety of the plant.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(f) EDCR 87-409, "Modifications of High Speed Valve Actuators", was completed April 5, 1989.

GENERAL SUMMARY:

Many Motor-Operated Valves (MOVs) at Vermont Yankee are required to change position quickly to allow particular modes of system operations. As the valve stem and disk are traveling and approach the open or closed seat, a torque switch or limit switch will actuate to interrupt motor current and stop valve travel. Due to the high stem speed of these valves, significant valve stem travel occurs during the period between switch contact opening and the cessation of motor rotation.

This EDCR was initiated to modify valves to reduce excessive seating thrust by addition of a mechanism to absorb overthrust. This reduced chronic valve damage and improved reliability. This modification reduced the possibility that the valve will reopen under DP conditions.

Control circuit changes (open on limit) were incorporated to reduce the overthrust during the open stroke. However, based on the need to ensure a tight seal while closing and the relative slow speed of the electrical switches, it was determined that a control circuit modification would not properly address the overthrust concern in the close direction.

In order to alleviate the problem of excessive overthrusting in high speed valves, Limitorque SB/SBD type operators were installed. The SB/SBD operator uses a Belleville spring pack assembly mounted on the top of the operator and is designed to limit the total seating thrust that can be delivered to the valve stem.

SAFETY EVALUATION SUMMARY:

There were three modifications contained in this EDCR. Some of the MOVs affected by this design change were converted from SMB to SB operators via the addition of a spring pack conversion kit. Secondly, the electromagnetic brakes were disabled and brake internals removed for 13 MOVs. Motor brakes were added to two MOVs.

The motor brakes on the 13 operators are not required to perform a specific safety function since the valve stem threads and/or operator gear sets are self-locking. The only safety concern is that a failure of the brake could impair the ability of the operator and valve to perform their respective safety functions.

The possibility of brake failure preventing valve movement will be eliminated by removing the internal parts of the brake and leaving only the housing to act as the motor end bell cover. The removal of the brakes will either reduce or not affect any overthrust condition.

For those valves which have nonlocking stem threads and whose operators have nonlocking gear sets, a mechanism is required to perform the safety function of locking up the valve stem to ensure the DP/system vibration cannot cause the valve to reopen. As a temporary measure, the motor brakes, which are certified as Class 1E and environmentally qualified in QDR 3.1, will perform this function.

The major component of this conversion is the addition of a spring pack to the drive sleeve assembly of the operator. This modification will allow the stem nut to float up (compressing the springs) in order to reduce total seating thrust values.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report and there is reasonable assurance that the health and safety of the public was not endangered.

(g) EDCR 87-403, "Reactor Vessel Water Level Reference Leg Temperature Monitoring Upgrade", was completed April 3, 1989.

GENERAL SUMMARY:

The Reactor Water Level Monitoring System (RWLMS) uses a reference leg as an integral part of its operation. When the RWLMS is calibrated, an assumed temperature is attributed to the reference leg for calibration purposes. If reference leg temperature differs from that assumed for calibration, an error will exist in the RWLMS

Furthermore, upon a rapid depressurization of the vessel, coincident with a high drywell temperature, the reference legs could approach a saturation condition. If saturation is achieved, the reference leg inventory would be diminished. This would result in errors in the RWLMS.

The Operator requires an indication of reliable reference leg temperature to determine the condition of the reference legs and, therefore, the degree of reliability of the RWLMS. This temperature information supports his efforts in using the EOPs as it will inform him of how reliable his RWLMS is.

Thermocouples installed on the reference legs provide the Control Room operator with reference leg temperatures. Thus, the operator is provided with the ability to determine when the Reactor Water Level Monitoring System may be compromised. By comparing these temperatures to vessel pressure, the operator knows the degree of error existing in the RWLMS.

SAFETY EVALUATION SUMMARY:

The modifications completed by this EDCR enhance the overall ability of the operator to determine the accuracy of the reference leg temperatures and, by extension, the Reactor Vessel Water Level Monitoring system to perform its function. This change is an improvement to safety because the procurement, installation, and documentation satisfies the criteria of IEEE 323-1974/DOR Guidelines, whereas the existing system did not.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(h) EDCR 86-412, "High Density Spent Fuel Racks", was completed July 14, 1989.

GENERAL SUMMARY:

The Vermont Yankee spent fuel storage pool was originally designed and licensed on the basis that a fuel cycle would be in existence that would only require storage of spent fuel for a year or two prior to shipment to a reprocessing facility. As the reactor core at Vermont Yankee contains 368 fuel assemblies with approximately 92 being replaced on an annual refueling schedule, a fuel storage capacity of 600 assemblies was considered adequate.

Once it became clear that reprocessing would not be available, Vermont Yankee took steps to provide additional on-site storage. In September 1977, Amendment No. 37 to the Vermont Yankee Operating License was granted by the NRC allowing installation of new racks to accommodate 2,000 spent fuel assemblies. This would permit Vermont Yankee to operate and maintain full core reserve discharge capability until 1990. However, government sponsored away-from-reactor storage is, and will be, unavailable until the late 1990's at the earliest. The capacity of the three(3) remaining PAR racks which can be installed per Amendment No. 37 is only good until 1990, therefore, an alternative for future fuel storage is necessary.

The Vermont Yankee Spent Fuel Storage Rack Replacement Report discussed several methods for on-site storage which were considered. However, it was concluded that the only practical alternative for Vermont Yankee for increasing on-site storage was by replacing the existing freestanding racks with a similar, proven design to allow closer spacing of the fuel assemblies. Therefore, this design change increased Vermont Yankee's storage capabilities by installing new stainless steel, freestanding high density storage racks containing a neutron absorbing material (Boral) in the existing pool.

To maximize the number of storage cells which could be installed in the existing pool, a number of pool modifications were included in this design change.

SAFETY EVALUATION SUMMARY:

The current design for the spent fuel pool calls for the installation of the three remaining PAR spent fuel racks which would bring the total number of spent fuel storage locations in the fuel pool to 1,990. However, this design change is substituting new stainless steel high density (NES) spent fuel racks for the PAR racks. This installation will provide for storage of up to 2,683 assemblies.

The new NES spent fuel racks have a nominal center-to-center spacing of 6.218 inches and also use Boral poison sheets surrounding each assembly for criticality control. A criticality analysis has been performed using proven benchmarked computer codes to show that the design of the NES racks provides for a multiplication factor (K-effective) less than the 0.95 limit for both normal and abnormal conditions.

Additionally, an evaluation was performed to determine the effect of the NES racks and PAR racks being simultaneously used to store fuel during installation of NES racks. This evaluation showed that if a minimum of 1.0 inch is maintained between NES racks and adjacent PAR racks after the maximum slide of each, the resultant K-effective is maintained below the 0.95 limit.

The NES racks are made of Type 304L stainless steel and utilize vented Boral canisters, that is, the Boral is subjected to the pool water. Numerous studies have been done on the performance of stainless steels in spent fuel storage pools, and their performance has been excellent. Stainless steel is used almost universally as the liners in the storage pools. The steel to be used in the fabrication of the NES racks is Type 304L, a low-carbon steel which is less susceptible to intergranual stress corrosion. Additionally, intergranual corrosion tests have been performed for all austenitic stainless steel used for the fabrication of the NES racks in accordance with ASTM A-262, Practice E. Considering the relatively benign environment and low temperatures in the Vermont Yankee spent fuel pool, the corrosion expected over the lifetime of the NES racks is negligible.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(i) EDCR 86-411, "Main Control Board Surface Modifications", was completed April 3, 1989.

GENERAL SUMMARY:

This EDCR addressed identified Human Engineering Discrepancies (HEDs) which the NRC had expressed concern on. The HEDs involved the repair of openings in the Control Room panels which resulted from removed equipment. Other modifications were included to improve the interface between the operators and the control panels. The modifications were divided into seven categories:

1) All openings in the control panels which resulted from removed equipment were sealed in a consistent manner. The openings had previously been sealed in a variety of ways. The new sealing method filled the holes from the inside of the control panel to achieve a flush surface on the front. This new method did not apply to locations where new equipment had been installed and support was required.

2) Existing switch escutcheon plates were replaced to improve the correlation between the switch position and the control function being performed. This included 11 escutcheon plates on CRP 9-6 and 3 escutcheon plates on CRP 9-23.

3) As a result of the extensive modifications to the main control board surfaces, some repainting was required. All six front panels were repainted in their entirety. The remaining panels were also repainted in their entirety.

4) Lines of demarcation were added to the main control board by painting 1/16" wide black lines in the areas identified. A total of 11 outlines, 28 brackets, and 1 division line were added to the panels.

5) All existing mimic lines were replaced to comply with the new color standard. This was a one-for-one replacement except for specific modifications on CRPs 9-3, 9-6, and 9-50.

6) Two hundred and fifteen (215) switch handles were replaced to meet the Vermont Yankee color standard. All remaining switch handles were to be black, and were replaced or painted if they were not already black.

7) All nameplates on the surfaces of the main control boards were replaced. There were approximately 3,200 nameplates. Forty percent of the nameplates were on the front panels. In addition, 31 throttle valves have been identified by affixing a letter "T" to the handle of the control switch.

SAFETY EVALUATION SUMMARY:

The escutcheon plates, nameplates, mimic, and paint were added to the safety class electrical control panels. The control switch handles are considered parts of the switches. Therefore, they have the same safety classifications as the switches themselves.

This EDCR had no direct effect on the manner in which any safety class or non-safety class equipment functions. As such, the probabilities associated with equipment failures were not altered. This EDCR modified the surface of the main control boards to implement the Vermont Yankee color and labeling standards. These standards have been applied consistently with the guidelines of NUREG-0700. These modifications reduce potential operator confusion and improve responsiveness to plant conditions. Any changes to the identification of Control Room equipment was consistent with the plant's operating procedures. The operators have been trained to use these procedures. Operator training improved the interface between man and machine, which reduces the probability of human error. Therefore, this EDCR indirectly improved the operation of the main control boards, which are safety class equipment. In addition, independent verification of the installation of all nameplates ensured that no items were incorrectly labeled. This verification eliminated all concerns that safety class equipment could be incorrectly labeled, potentially leading to operation problems. The functions and operations of all plant systems remain unchanged. Therefore, new errors which can be associated with system modifications were not introduced by this design change. An analysis of these modifications has determined that there are no adverse effects on the structural integrity of the MCB.

This design change did not present significant hazards not described or implicit in the safety analysis report, and there is reasonable assurance that the health and safety of the public was not endangered.

(j) EDCR 86-408 ECN1, "CRDR Modifications to CRP 9-3 through CRP 9-8", was completed April 4, 1989.

GENERAL SUMMARY:

As a result of NUREG-0737, Vermont Yankee completed its Detailed Control Room Design Review (DCRDR) and submitted a summary report to the NRC. The DCRDR resulted in certain Human Engineering Discrepancies (HEDs).

Implementation of this design change did not add new or additional capability to the plant. However, the change improved the man machine interface in the control Room by resolving the HEDs as follows:

1) This HED involved the relocation of the 345 KV digital voltmeter from CRP 9-8 to CRP 9-7. The purpose of the move was to place the indicator in close proximity with the AC voltage adjustment control. Prior to the design change, the indicator was located in CRP 9-8 and the adjustment control was on the CRP 9-7 bench board.

2) This HED involved intermixed vessel level and pressure indicators which were difficult to read. Prior to the design change, the CRP 9-5 reactor vessel level and pressure indicators were Sigma dual indicators and were intermixed with other single reactor level indicators. The modification involved the removal of the two dual indicators which were replaced with four single indicators. The indicators were rearranged, as well as the recorders mounted above them, into functional groupings to facilitate comparative readings.

3) This HED involved the annunciator response controls. The existing Annunciator System was split into two response zones, one for NSSS and one for BOP. The HED resolution split the response controls into four zones: CRP 9-8, CRPs 9-7 and 9-6, CRPs 9-5 and 9-4, and one for CRP 9-3.

   The annunciator controls were arranged in a standard configuration at each response location.

   In addition, the annunciator controls were located at the same relative position on each control panel.

   The combination of changes described above ensures the best possible combination of operator convenience and assurance that all alarms will be properly acknowledged.

SAFETY EVALUATION SUMMARY:

This change involved both Safety Class Electrical (SCE) and Non-Nuclear Safety (NNS) components and Systems.

All of the equipment is being relocated in Control Room panels which are themselves safety class.

Implementation of this change did not result in any change to the Technical Specifications, plant operations, protective functions, or design bases of the plant.

Implementation of this change improved the manner in which this equipment operates. The operators were properly trained to familiarize themselves with this change.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(k) EDCR 86-407, "CRP 9-2 Modifications", was completed March 29, 1989.

GENERAL SUMMARY:

Implementation of this design change resulted in the relocation and/or removal of equipment in order to: (a) improve correlation between indicating and back-up indicating devices on CRP 9-2, (b) aid operator action by making it easier to locate devices and to depict devices in a manner which minimizes confusion, and (c) ensure consistency on Control Room panels. The implementation of this design change only changed the location of equipment; operation of all relocated equipment remained the same. Equipment to be removed was either 1) existing equipment which had previously been disconnected and made nonfunctional or 2) equipment which was no longer required.

SAFETY EVALUATION SUMMARY:

Relocation and rewiring of monitors, recorders, and controllers on CRP 9-2 did not change the operation or function of any of this equipment. This equipment was relocated to resolve a HEDS identified during the DCRDR performed in accordance with NUREG-0700, "Guidelines for Control Room Design Reviews".

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(i) PDCR 88-06, "Appendix R Cable Tray Fire Stops", was completed November 29, 1989.

GENERAL SUMMARY:

A requirement of Section III.G of Appendix R to 10 CFR 50 is that one train of systems necessary to achieve and maintain hot shutdown conditions from either the Control Room or Emergency Control Stations is free of fire damage. Vermont Yankee, as discussed in the Fire Hazards Analysis, considers the Reactor Building to be technically one large fire area but separated into distinct zones by floors, walls and inherent spatial separation.

Fire stopping cable trays prevents the propagation of fire horizontally along a cable tray, and helps establish corridors of "no intervening combustibles" within the Reactor Building.

During the 1986 refueling outage, fire stopping material was added to a number of cable trays by EDCR 85-402. This design change installed fire retardant material at each end of the 20 foot separation zones. Although this installation was earlier viewed by the NRC as acceptable, it did not meet the current interpretation of Appendix R requirements. This interpretation was presented during an NRC Appendix R compliance audit in 1988. The audit recommendations concluded that the fire retardant material should be applied over the entire 20 foot separation zone.

PDCR 88-06 provided for the installation of fire retardant material in cable trays within the entire 20 foot separation zones.

SAFETY EVALUATION SUMMARY:

The implementation of this design change ensured the continued availability of safe shutdown systems in the event of a fire. The installation did not change any existing equipment configuration or operation.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(m) PDCR 88-03, "Recirc Pump Motor Snubber Removal", was completed February 25, 1989.

GENERAL SUMMARY:

This PDCR eliminated the four 100 KIP snubbers installed on the recirculation pump motors during the Recirc Pipe Replacement Outage per EDCR 85-01. General Electric's ensuing as-built pipe stress analysis was completed and found to be acceptable with the absence of the four motor snubbers. The resultant effect is a reduction of maintenance, personnel exposure and elimination of off-site equipment testing.

SAFETY EVALUATION SUMMARY:

The design basis seismic inputs for the as-built stress analysis are equivalent to that used in EDCR 85-01, are more conservative than that of the original design, and are in accordance with the Vermont Yankee seismic reanalysis program as approved by the USNRC. Because the as-built analysis utilized current industry codes and USNRC seismic criteria, elimination of the SS-4 snubbers complies with current amendments to General Design Criterion 4 of 10 CFR 50, Appendix A (as referred to in a published Commission response to Issue 16 in the Federal Register on October 27, 1987).

This alteration did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(n) PDCR 88-02, "Reactor Building Air Temperature - Torus Area Level & Drywell Pressure Indication (CRDR)", was completed April 3, 1989.

GENERAL SUMMARY:

On January, 12, 1987 a DCRDR Implementation Task Force meeting was held to discuss "EOP Task Analysis" and "Dynamic EOP Evaluation". The result of this meeting was the identification of the following hardware changes needed for proper execution of Emergency Operating Procedure OE 3105, "Secondary Containment Control Procedure".

1) Addition of eight Reactor Building area temperature monitors.

2) Addition of Reactor Building "Max. Norm. Oper. Level" and "Max. Safe Oper. Level" indication.

3) Conversion of present drywell/torus psia pressure indicators and recorders to read out and record in psig.

Each of these changes was completed as described.

SAFETY EVALUATION SUMMARY:

The Control Room indication added by this PDCR is classified as nuclear non-safety (NNS) with the exception of the replacement Control Room indication scales which are safety class. These were replaced with safety class scales of the same type and weight and can, therefore, be considered a one-for-one replacement. Therefore, this change did not degrade the function or seismic qualities of the devices or cabinet.

All changes to the main control board by this PDCR have been determined to present negligible changes in weights when compared with the overall weight of the equipment and do not affect stiffness of the panel, thereby creating no change in the seismic response.

This alteration did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(o) PDCR 88-01, "Reactor Building Load Shed Relays", was completed March 30, 1989.

GENERAL SUMMARY:

This PDCR implemented changes to the Load Shedding Relays for MCC 8B and MCC 9B. Because of EQ concerns, the GE HFA Load Shedding Relays 94/8B, 94/9B and 94/9B-1 for MCC 8B and 9B were removed and Weidmuller terminal blocks were installed in the same location. The Load Shedding function is now accomplished utilizing spare contacts on the 94/8 and 94/9 relays in Bus 8 and Bus 9. This action removed the HFA relays from the harsh environment of the Reactor Building. The 94/8 and 94/9 relays are now located in the Switchgear Room which is a mild environment.

SAFETY EVALUATION SUMMARY:

This design removed the local HFA relays in the MCC's and replaced them with qualified terminal blocks that are intended for the Reactor Buildings environmental requirements. This modification restored the load shedding function to MCC 8B and MCC 9B via the HFA relay in the 480 volt switchgear Bus 8 and Bus 9 located in a mild environment. The addition of the two fuses in the 480 volt switchgear enhances the reliability of the load shedding scheme by preventing a fault in the shunt trip coils or the feeder cable from disabling the remaining shunt trip system.

This alteration did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(p) PDCR 86-07, "Scram Lights/DG VAR Meter/HPCI Start - CRDR 3", was completed April 3, 1989.

GENERAL SUMMARY:

This PDCR is a result of Vermont Yankee Commitments to the NRC to correct Control Room human engineering deficiencies stemming from the Detailed Control Room Design Review Program Plan. The program plan evolved from several studies which were performed in an effort to meet NRC regulatory requirements which are defined in Generic Letter 82-33 and NUREG-0737, Supplement 1. The objective of the program plan was to review and improve, where necessary, the man machine interface in the station Control Room to promote safe and efficient plant operation. This PDCR implemented specific portions of three (3) human engineering discrepancy (HED) findings resultant from the review. The HEDs and their associated resolutions are listed as follows:

1) HED 1405 - Add Scram Solenoid Group Lights on CRP 9-5

2) HED 1405 - Add Diesel Generator VAR Meters on CRP 9-8

3) HED 0602 - Add HPCI System Ready Light on CRP 9-3

4) HED 05B0 - Add MTS-2 Trip Light on CRP 9-7

SAFETY EVALUATION SUMMARY:

The addition of VAR meters on CRP 9-8 enhances the operator's ability to remotely regulate diesel generator loading. The addition of scram lights on CRP 9-5 provides operators with immediate visual indication of scram bus activity. The addition of a HPCI System Ready Lamp on CRP 9-3 enables operators to quickly determine HPCI system status. The addition of a MTS-2 lamp on CRP 9-7 provides the operators with a second means of verifying MTS-2 status in the event of a trip light failure. Therefore, these additions enhance the efficiency and overall safety of the plant.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(q) PDCR 86-006, "Alarms/Lights/Indicators - CRDR 2", was completed April 3, 1989.

GENERAL SUMMARY:

This PDCR is a result of Vermont Yankee commitments to the NRC to correct Control Room human engineering deficiencies (HED's) resulting from the Detailed Control Room Design Review Program (DCRDR) Plan. The program plan evolved from several studies which were performed in an effort to meet NRC regulatory requirements, which are defined in Generic Letter 82-33 and NUREG-0737, Supp. 1. The objective of the plan was to review and improve, where necessary, the man machine interface in the station Control Room to promote safe and efficient plant operation. This PDCR implemented specific portions of one (1) HED and two (2) resultant findings from the review. The HED and its' associated findings are listed as follows:

1) HED 0200, Finding 0407 - Align indicating lights to facilitate comparative readings for specific radiation monitors on CRP 9-10, 12.

2) HED 0200, Finding 0435 - Align indicating lights to facilitate comparative readings for the Outdoor Page Silence switch, the Station Air Compressor Cooling Water Transfer switch, and the Iodine Filter switch located on CRP 9-23.

SAFETY EVALUATION SUMMARY:

The modification to the Main Steam Radiation Monitors is Safety Class Electrical, all other modifications are Non-Nuclear Safety. Portions of the Control Room panels, where the equipment affected by this design change are located, are both Safety Class and Non-Nuclear Safety as discussed in Section 5.

The modifications implemented by PDCR 86-006 are a result of a DCRDR which was performed in accordance with NUREG-0700, "Guidelines for Control Room Design Reviews", and which was coordinated with the development of Symptom Based Emergency Operating Procedures. This design change resolved HEDs by arranging indicating lights in a consistent manner. As part of Vermont Yankee's ongoing Operator Training Program, any change(s) to equipment which alters its operational characteristics is included in future training programs.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(r) PDCR 86-05, "Switch Relocation (CRDR-1)", was completed March 29, 1989.

GENERAL SUMMARY:

The PDCR is a result of Vermont Yankee commitments to the NRC to correct Control Room Human Engineering deficiencies stemming from the Detailed Control Room Design Review (DCRDR) Program. The program plan evolved from several studies which were performed in an effort to meet NRC regulatory requirements which are defined in Generic Letter 82-33 and NUREG-0737, Supplement 1. The objective of the program plan was to review and improve, where necessary, the man machine interface in the station Control Room to promote safe and efficient plant operation. This PDCR implemented specific portions of four (4) Human Engineering Discrepancy (HED) findings resultant from the review. The HEDs and their associated resolutions are listed as follows:

1) HED 0102 - Relocate Condenser Vacuum Isolation Bypass Switches on CRP 9-15/9-17

2) HED 05C0 - Rewire Circ Wtr Booster Pump Bypass Gate Control Switches on CRP 9-6

3) HED 05E0 - Revise keylock switch directions for the following switches:

   RHRSW Pump A/C & B/D Manual Override Switches on CRP 9-3

   Core Spray Sys 1/2 Drywell Pressure Test Switches on CRP 9-3

4) HED 0601 - Eliminate the RHR Head Spray Valve Control Switches and Indicators on CRP 9-3

SAFETY EVALUATION SUMMARY:

The modifications to the Core Spray, RHR, and Condenser Isolation Bypass switches are classified as Safety Class Electrical. All other modifications that affect Control Room panels are classified as Non-Nuclear Safety.

The modifications implemented by PDCR 86-05 are a result of a DCRDR which was performed in accordance with NUREG-0700, "Guidelines for Control Room Design Reviews", and which was coordinated with the development of Symptom Based Emergency Operating Procedures. This design change resolved HEDs by rearranging and labeling control switches in a manner consistent with other similar switch types. As part of Vermont Yankee's ongoing operator training program, any change(s) to equipment which alters its' operational characteristics is included in future training programs.

This design change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(s) PAR 88-19, "Installation of Flow Meters On the Potable Water System", was completed January 13, 1989.

GENERAL SUMMARY:

This PAR installed water meters on the potable water system. These meters were installed to provide a means of quantifying water consumption. The meters will normally be in service. Each of the meters that were installed was provided with isolation valves that allow removal of the flow meter without loss of the water supply. Strainers were provided for all of the meters, except the meter on the potable water tank supply line. Sufficient filters exist for this installation.

SAFETY EVALUATION SUMMARY:

This PAR added water meters to the potable water system. This provided the capability to quantify water usage from the various plant wells and identify the specific usage for the various plant locations that are supplied with potable water. This PAR did not affect the operation of any safety systems.

This alteration did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(t) PAR 88-06, "Stack Gas Sample Point", was completed January 23, 1989.

GENERAL SUMMARY:

This PAR provided the appropriate documentation to permanently install Mechanical Bypass 86-0040. The Mechanical Bypass was installed to relocate the Stack Grab Sample System sample location from the suction side of the vacuum pump to the discharge side of the vacuum pump. The grab sample is taken using a plastic sample bottle. Moving the sample location to the discharge side of the vacuum pump ensured that the volume of the bottle is not affected by the negative pressure on the suction side. One of the valves previously used for stack sampling (SRS 22) was removed from the stack sampling system. The tygon tubing in place between valves SRS 21 & 22 was replaced by stainless steel tubing. The mechanical bypass had been in place for more than ten months, and had performed its' function adequately.

SAFETY EVALUATION SUMMARY:

Although the stack sample point system is illustrated in the FSAR, it is a non-safety system and is not included in the FSAR text.

Performance of the Stack Sample Point System was not affected by the installation of this PAR. The sample is obtained in the same manner as before the installation of MBR 86-0040, simply from a different location.

This alteration did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(v) PAR 86-02 Supplement 1, "Transverse Incore Probe (TIP) System", was completed January 23, 1989.

GENERAL SUMMARY:

This supplement to PAR 86-02 provided an environmentally qualified interface for the TIP ball valves to prevent the valves from opening after a LOCA due to an environmentally induced failure. It also incorporated all changes made by GE during the installation of the system.

During installation, the concern was raised that since relays K3, K4, and the associated internal wiring in the TIP drive mechanism were not environmentally qualified that they could fail after a LOCA causing the ball valves to reopen. To prevent this, K3 and K4 were replaced with environmentally qualified relays, and were relocated to a location separate from the drive mechanism.

To accomplish this, terminal boxes were installed next to the conduits containing the control cables to the drives to allow separating the control for K3 and K4 relays from the cable ahead of the drive mechanism. This prevents any failure in the drive mechanism from affecting K3 and K4 relays. Since all the conductors in the control cables are not used in the new drives, only those required were terminated in the boxes and connected to the drive mechanisms.

The electronics for the proximity sensors were removed from box 417, as they are not used with the new system. In addition, the new system installed an electrical stop to prevent inadvertent TIP withdrawal outside of the TIP Room.

SAFETY EVALUATION SUMMARY:

The TIP system, with the exception of the ball and shear valves which were not changed, is classified as non-nuclear safety.

The system functions the same as the original system to perform the containment isolation functions.

This change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(v) Temporary Modification 89-042, was implemented July 7, 1989 and restored October 4, 1989.

GENERAL SUMMARY:

This Temporary Modification installed equipment for the detection of leakage within the Off-Gas Sampling System. Helium injection equipment was attached to the Off-Gas Sampling System. Helium was then injected into the system. The test equipment was then utilized to located and monitor any system leakage.

SAFETY EVALUATION SUMMARY:

The helium detection equipment was connected to sampling inlet and outlet valves in the existing system. The respective valves were closed during installation and removal of this modification. This modification did not, in any way, hinder the operation of the Off-Gas Monitoring System. The materials used were determined to be acceptable for this application.

This Temporary Modification did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(x) Temporary Modification 89-035 was implemented June 26, 1989 and restored June 26, 1989.

GENERAL SUMMARY:

This Temporary Modification installed equipment for the detection of leakage within the Off-Gas Sampling System. Helium injection equipment was attached to the Off-Gas Sampling System. Helium was then injected into the system. The test equipment was then utilized to locate and monitor any system leakage.

SAFETY EVALUATION SUMMARY:

The helium detection equipment was connected to sampling inlet and outlet valves in the existing system. The respective valves were closed during installation and removal of this modification. This modification did not, in any way, hinder the operation of the Off-Gas Monitoring System. The materials used were determined to be acceptable for this application.

This Temporary Modification did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(y) Temporary Modification 89-030 was implemented May 25, 1989 and restored June 5, 1989.

GENERAL SUMMARY:

This Temporary Modification provided a means to transfer the passive seal from the Reactor Building equipment airlock inner door to the outer door. This action was necessary to support scheduled work dealing with the High Density Spent Fuel Racks Project. This modification ensured that Secondary Containment was maintained. Following the Fuel Rack work the passive door seal was restored to its' original configuration.

SAFETY EVALUATION SUMMARY:

There was reasonable assurance due to the controls established that the transfer of the passive door seal material from the inner truck lock door to the outer truck lock door would provide the necessary personnel safety and did not affect the integrity of the Secondary Containment.

This Temporary Modification did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(z) Temporary Modification 89-029 was implemented May 15, 1989 and restored May 25, 1989.

GENERAL SUMMARY:

This Temporary Modification provided a means to transfer the passive seal from the Reactor Building equipment airlock inner door to the outer door. This action was necessary to support scheduled work dealing with the High Density Spent Fuel Racks Project. This modification ensured that Secondary Containment was maintained. Following the Fuel Rack work the passive door seal was restored to its' original configuration.

SAFETY EVALUATION SUMMARY:

There was reasonable assurance due to the controls established that the transfer of the passive door seal material from the inner truck lock door to the outer truck lock door would provide the necessary personnel safety and did not affect the integrity of the Secondary Containment.

This Temporary Modification did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(aa) Temporary Modification 89-012 was implemented August 16, 1989 and restored September 14, 1989.

GENERAL SUMMARY:

This Modification installed 3/4" stainless steel tubing on the HPCI Gland Seal Exhauster pump casing drain. The pump casing drain plug was replaced with a 1" NPT to 3/4" tubing adapter and the tubing was field run to the nearest floor drain. This drain line prevented excessive moisture collection in the pump which could have affected performance. A loop seal was placed in the line to prevent air entrainment during pump operation and airborne releases through the line during pump standby.

SAFETY EVALUATION SUMMARY:

Installation of the drain line did not affect the operation or efficiency of the Gland Seal Exhauster or the HPCI system as a whole. During operation, the loop seal prevented air entrainment within the pump.

This Temporary Modification did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(bb) Temporary Modification 89-002 was implemented April 27, 1989 and restored May 3, 1989.

GENERAL SUMMARY:

This Temporary Modification removed three sample steam lines and one isolation valve from the Main Steam Sample System. The modification consolidated four sample paths to a single sampling path.

SAFETY EVALUATION SUMMARY:

The reduction in the number of lines running from the Main Steam lines in the Turbine Building to the Sample Sink Panel reduces the probability of accident occurrence since there are fewer components and connections under pressure and temperature conditions. The added isolation valve and tubing hardware are all equivalent to that required by the original design specification.

This Temporary Modification did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(cc) Mechanical Bypass 89-0027 was installed April 8, 1989 and removed April 26, 1989.

GENERAL SUMMARY:

This Temporary Mechanical Bypass installed a drain line from a 55 gallon collection drum to the Reactor Building Roof Drain System. A submersible pump was installed in the collection drum with a flexible hose discharging to the Reactor Building Drain System. The Roof Drain System, at this location, is approximately 20 feet higher than the collection drum.

A 9 inch loop seal with 2 check valves was installed in the discharge line to prevent backflow from the roof drain.

SAFETY EVALUATION SUMMARY:

The primary concern with this Mechanical Bypass was Secondary Containment integrity. MBR 89-0008 provided two seals which maintained Secondary Containment.

1) The submersible pump was maintained under water whenever the roof drain valve was open.

2) The loop seal was filled with water whenever the roof drain was opened.

The water seals were maintained prior to, and throughout the time that Secondary Containment integrity was not maintained by the roof drain isolation valve.

This activity did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(dd) Mechanical Bypass 89-0009 was installed February 16, 1989 and removed March 2, 1989.

GENERAL SUMMARY:

This temporary Mechanical Bypass provided a means to transfer the passive seal from the Reactor Building equipment airlock inner door to the outer door. This action was necessary to support scheduled work dealing with the High Density Spent Fuel Racks Project. This modification ensured that Secondary Containment was maintained. Following the Fuel Rack work the passive door seal was restored to its' original configuration.

SAFETY EVALUATION SUMMARY:

There was reasonable assurance due to the controls established that the transfer of the passive door seal material from the inner truck lock door to the outer truck lock door would provide the necessary personnel safety and did not affect the integrity of the Secondary Containment.

This temporary Mechanical Bypass did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(ee) Mechanical Bypass 89-0008 was installed February 14, 1988 and removed February 22, 1989.

GENERAL SUMMARY:

This temporary Mechanical Bypass installed a drain line from a 55 gallon collection drum to the Reactor Building Roof Drain System. A submersible pump was installed in the collection drum with a flexible hose discharging to the Reactor Building Drain System. The Roof Drain System, at this location, is approximately 20 feet higher than the collection drum.

A 9 inch loop seal with two check valves was installed in the discharge line to prevent backflow from the roof drain.

SAFETY EVALUATION SUMMARY:

The primary concern with this Mechanical Bypass was Secondary Containment integrity. MBR 89-0008 provided 2 seals which maintained Secondary Containment.

1) The submersible pump was maintained under water whenever the roof drain valve was open.

2) The loop seal was filled with water whenever the roof drain valve was opened.

The water seals were maintained prior to, and throughout the time that Secondary Containment integrity was not maintained by the roof drain isolation valve.

This activity did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(ff) Mechanical Bypass 88-0052 was installed February 15, 1989 and removed May 3, 1989.

GENERAL SUMMARY:

This Temporary Mechanical Bypass installed 2 Atlas Copco (AC) PTS 1200 oil-free air compressors into the Service Air System. The installation of the external air compressors allowed the Service Air System to provide air while the Service Water System, which supplied cooling water to the station air compressors, was being worked on during the outage.

SAFETY EVALUATION SUMMARY:

The AC compressors were connected via hoses to the discharge piping of Station Air Compressor C-1-1d. If the hose was punctured or sheared, the failure of check valve V72-1d to function, could cause a loss of service and instrument air header pressure. To ensure that valve V72-1d functioned correctly, MBR installation details required that the check valve be internally inspected and leak tested.

MBR 88-0052 affected the Service Air System. The Service Air System supplies instrument air. Instrument air supplies many safety related components as well as having the ability of requiring the Reactor to be scrammed on low scram air header pressure. A review of this MBR identified two potential interface effects: excessive or loss of air header pressure. The design of the MBR, as well as the fact that this MBR has been successfully used in the past (MBR 87-0014), ensured that this system would operate normally with the MBR installed.

MBR 88-0052 did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health of the public was not endangered.

(gg) Mechanical Bypass 88-0032 was installed October 12, 1988 and removed January 26, 1989.

GENERAL SUMMARY:

This Temporary Mechanical Bypass provided the RBCCW Surge Tank with a level indication. Tygon tubing was installed at RCW-907 valve and extended vertically up the side of the tank and left open to the atmosphere.

SAFETY EVALUATION SUMMARY:

Since the Surge Tank and its' attachments were Safety Class 3, continuous level indication through the open RCW-907 valve required the existing Tygon to be Safety Class 3 with seismic mounting.

Tubing was installed such that it extended to an elevation higher than the Surge Tank high level alarm elevation. This eliminated possible tubing leakage if the tank was overfilled and RCW-907 was then opened.

This temporary Mechanical Bypass did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(hh) Mechanical Bypass 88-0030 was installed October 13, 1988 and removed January 12, 1989.

GENERAL SUMMARY:

This temporary Mechanical Bypass provided the TBCCW Surge Tank with a level indication. Tygon tubing was installed at TCW-832 valve and extended vertically up the side of the tank and left open to the atmosphere.

SAFETY EVALUATION SUMMARY:

The installation of the tubing had no affect on the TBCCW systems' operation.

In the event of tubing failure, the Demineralized Water Transfer System would have provided leakage make-up to the Surge Tank. This leakage would have been detected on AO rounds. If leakage flow had exceeded make-up, the TBCCW Surge Tank low level alarm would annunciate in the Control Room to identify the problem. This would have initiated immediate AO response to the area.

This Temporary Mechanical Bypass did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(ii) Mechanical Bypass 88-0023 was installed August 5, 1988 and removed January 31, 1989.

GENERAL SUMMARY:

Mechanical Bypass 88-0023 temporarily provided approximately 10 gpm to 50 gpm (max) of service water to a temporary cooling fan for the service water pump motor P-7-1a. All of the components used were of greater or equal pressure rating compared to the already existing piping.

SAFETY EVALUATION SUMMARY:

MBR 88-0023 did not significantly impact the Service Water (SW) System.

The materials of construction had a higher pressure rating than the SW system and therefore were acceptable.

There were no affects on plant operation while the bypass was installed. This activity did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(jj) Mechanical Bypass 86-0039 was installed October 14, 1986 and removed April 3, 1989.

GENERAL SUMMARY:

This Temporary Mechanical Bypass installed a Demineralizer Skid into the MUD (Make-up Demineralizer) System, which provided an equivalent (or better) water quality compared to the existing system.

SAFETY EVALUATION SUMMARY:

MBR 86-0039 did not tie directly into a safety class system. The change was located such that its' mechanical failure would not affect a safety class system. The material of this bypass was evaluated to be acceptable for both chemical and mechanical service.

MBR 86-0039 did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(kk) LL/JR 89-0013 was installed on February 11, 1989 and removed February 15, 1989.

GENERAL SUMMARY:

This LL/JR inputed the rod "full-in" signal for control rod 22-11. The normal "full-in" rod position indication for rod 22-11 was not functioning. The green light did not light and therefore no rod other than 22-11 could have been moved due to the refuel interlocks. During refueling, no more than one control rod could be withdrawn from its' fully inserted position.

SAFETY EVALUATION SUMMARY:

This request did not involve any unreviewed safety questions based on:

1) Rod 22-11 was know to be fully inserted.

2) The LL/JR was only in effect while the directional control valves for 22-11 were disarmed electrically to prevent inadvertent withdrawal.

3) The Hydraulic Control Unit was white tagged & hydraulically locked by closing the 13-101 and 13-102 valves.

The intent of the refuel interlocks were therefore satisfied.

This change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(ll) LL/JR 88-0021 was installed July 26, 1988 and removed April 4, 1989.

GENERAL SUMMARY:

This LL/JR installed a relay annunciator circuit for the cooling water high leakage alarm for recirculation pump motor "A". This was done to isolate an intermediate ground in the signal cable C1706J from the DC annunciator system. The relay circuit utilized the existing level switch to operate an AC relay. The AC relay output contacts were then connected to the Control Room annunciator. The grounded lead of the signal cable was connected to the neutral side of the 120V AC supply. A power monitor relay was also provided in the relay circuit to alarm for any AC power loss in the relay circuit. Power for the relay circuit was from the AC outlet in CRP 9-4, fuse AA-f9(15A) LP-1L CKT #22.

SAFETY EVALUATION SUMMARY:

This LL/JR was not connected to any safety circuits. The alarm system was non-safety as were the recirc. pumps themselves, along with their power supplies.

The LL/JR enchanced operation by providing continued monitoring of the "A" Recirc. Pump Cooling Water Leakage Alarm that had the ground in the cable without degrading the Control Room annunciator system.

The changes provided operation in a manner shown in the FSAR and therefore did not present a significant hazard as described or implicit in the Safety Analysis Report and there was reasonable assurance that the health and safety of the public was not endangered.

(mm) LL/JR 87-0131 was installed September 23, 1987 and removed March 26, 1989.

GENERAL SUMMARY:

This LL/JR disconnected the HFA relays whose environmental qualifications were found to be indeterminate. Only the HFA relays in the MCC 8B and MCC 9B were of concern since they were the only load shedding relays in a harsh environment. Load shedding takes place following a loss of Normal Power (LNP) to limit the loading on the Diesel Generator (DG). With this modification, failure of the HFA relays in MCC 8B and MCC 9B, could not affect the operation of the other load shedding relays.

SAFETY EVALUATION SUMMARY:

The temporary change increased the reliability of the Electrical Distribution System by preventing a failure of the HFA relays in MCC 8B and MCC 9B from disabling other portions of the load shedding circuitry. Although load shedding on the MCCs were bypassed, the increased DG loading was negligible and all of the affected equipment still performed their safety functions.

Therefore, this change did not increase the probability or consequence of a previously analyzed accident. It does not create the possibility of a new type of accident and it does not reduce the margin of safety as defined in the basis of the Technical Specifications.

For the above reasons, the change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

(nn) LL/JR 87-0045 was installed September 18, 1987 and removed April 1, 1989.

GENERAL SUMMARY:

This LL/JR bypassed a portion of cable and a knife switch which were determined not environmentally qualified. The qualification status of the switch was determined to be indeterminate due to a lack of adequate qualification documentation. Qualification of the existing component was highly unlikely because the manufacturer could not be determined. Since this type of equipment was not normally used in applications requiring environmental qualification, obtaining a fully qualified replacement would have been difficult.

SAFETY EVALUATION SUMMARY:

The components affected by this change were Safety Class Electrical. By removing the transfer switch from service using the proposed LL/JR, operational flexibility was decreased in that the MCC by itself could not have been alternately fed from Bus DC-1.

However, if power to Bus DC-2 was lost, MCC-DC-2A as well as all other loads on DC-2 could be powered from DC-1 through the use of the Bus tie breakers. Because the proposed qualified splice was used to replace the existing transfer switch which lacked documentation of qualification, assurance of operation during accident scenarios was increased.

Since the ability of the MCC to perform its' safety function was not degraded, there was no increase in the probability or consequences of an accident.

This change did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

B. Tests & Experiments

1. Test Procedure, "Cable Vault Room Enclosure Integrity Test" was completed on November 2, 1989.

GENERAL SUMMARY:

This procedure provided a method to equate Cable Vault enclosure leakage to worst case Carbon Dioxide leakage. Enclosure leakage was determined by the tracer gas and door fan tests. The calculation method provided by NFPA 12A, App. B, 1989 edition makes it possible to predict the level of the descending interface of the CO_2/air mixture with respect to time. This calculated value provided a conservative prediction of the CO_2 concentration and maximum hold time for the Cable Vault Room.

The CO_2 total flooding system was installed in the Cable Vault to provide protection from a potential deep seated fire. The original installation was a manually activated system, which was installed in 1970. In 1977 the system was upgraded to an automatically initiated system with second shot capability provided from the West Switchgear Room. The system was not discharge tested for design concentration as such a test was not required by NFPA 12, 1968 or 1973 edition.

In 1977 a Tech. Spec. amendment was issued which reflected the modifications to the Cable Vault CO_2 System. The Safety Evaluation Report which was issued in support of this amendment, refers to the design criteria for the system. At the time of installation the system was tested to the criteria of the National Fire Protection Association standard for CO_2 systems (NFPA-12, 1977). The standard of record at that time did not require a full discharge test. Recently, the NRC interpreted that a full discharge test was required to prove the adequacy of the CO_2 system.

The NRC stated that they would entertain an alternate to the full discharge test if a conclusive test method could be designed.

To meet the NRC interpretation of NFPA 12, and to test the adequacy and operability of the CO_2 system, a tracer gas test and an Enclosure Integrity Test were completed in the Cable Vault Room. This verified that the Cable Vault CO_2 System was physically capable of providing and maintaining a CO_2 concentration of 50% CO_2 (by volume) for a period of ten minutes.

SAFETY EVALUATION SUMMARY:

The Cable Vault Automatic Total Flooding CO_2 Suppression System (mechanical equipment, piping, and electrical control equipment) is a non-safety class system. The CO_2 system provides fire protection for safety class electrical cable and equipment, and is therefore designated a Vital Fire Protection System. The walls which comprise the boundaries of the Cable Vault are designated as Vital Fire Barriers. These walls provide protection from fires which may occur outside of the Cable Vault. Additionally, the walls are designed to confine a fire which may occur within the Cable Vault. The walls also serve the function of confining the CO_2 which will be discharged in the event of a fire within the Cable Vault.

The discharge of CO_2 in the Cable Vault would raise a potential life safety consideration. The completion of this test procedure did not involve the actual discharge of CO_2.

The initial portion of this test involved the introduction of a tracer gas to the Cable Vault atmosphere. The tracer gas utilized in this test was Sulfur Hexafluoride (SF_6). The tracer gas was used in concentrations of 250 ppm or less. This level of SF_6 is well below the toxic limit of 1000 ppm.

The potential for overpressurization of the room due to the discharge of CO_2 was evaluated utilizing the guidance provided in NFPA No. 12; 1989. This possibility was examined through a comparison of the Equivalent Leakage Area as determined by this test with the required free venting area as determined by the NFPA 12 code requirements. This ensured that sufficient vent area is available to prevent excessive pressure buildup within the room during a CO_2 discharge.

The tracer gas and Enclosure Integrity test of the Cable Vault verified the design capabilities of the total flooding CO_2 system.

This test did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

C. Safety and Relief Valve Challenges and/or Failures

1. None

D. Special Test Procedures

1. Special Test Procedure 89-02, "CS-5A & CS-26A Data Acquisition" was completed on February 8, 1989.

GENERAL SUMMARY:

The purpose of this test was to gather information on valve thrust performance with an applied differential pressure. The information was gathered using the Motor-Operated Valve Analysis and Test System (MOVATS).

NRC IEB 85-03 requires the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems' Motor-Operated Valves (MOVs) to be reviewed and verified operable at any differential pressure the valve could be subjected to. This work was extended to include the Core Spray and RHR Systems.

IEB 85-03 requires as much correlation as possible between calculated thrust and the measured thrust needed during valve stroke under differential pressure. Not all MOVs can be stroked with an appropriate differential pressure. The differential pressure seen during accident or inadvertent valve operation may be much higher than can be achieved in a controlled test where prevention of system/component damage is mandatory.

Several valves were chosen for evaluation based on accessibility and available differential pressure. Core Spray valves CS-5A and 26A could be tested at any time, whether the plant was in operation or in shutdown mode. CS-5A is the minimum flow bypass valve for Core Spray Pump P-46-1A, while CS-26A is a full flow test valve for P-46-1A.

SAFETY EVALUATION SUMMARY:

The Core Spray System mechanical equipment and piping are Safety Class 2. The Core Spray System electrical control equipment is Safety Class Electrical.

The Core Spray System is an Emergency Core Cooling System (ECCS) which, in conjunction with the primary and secondary containments, has the safety objective of limiting the release of radioactive materials to the environs following a loss-of-coolant accident, so that resulting radiation exposures are kept to a practical minimum and are within the guideline values given in 10CFR100.

During the short periods in this special test procedure when CS-5A was de-energized to hook up and remove the MOVATS equipment, one loop of the Core Spray System was not able to perform its safety function. However, the other independent Core Spray loop and the RHR (LPCI) subsystems were available to fulfill the safety function of the Core Spray System.

Prior to starting this special test, the other independent core spray loop, LPCI and the diesel generators were demonstrated to be operable. This provided assurance that these systems, when used in concert with the primary and secondary containments, would satisfy the ECCS safety objective as stated in the FSAR.

This test did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

2. Special Test Procedure 89-01, "HPCI-14 Data Acquisition" was completed February 7, 1989.

GENERAL SUMMARY:

The purpose of this test was to gather information on valve thrust performance with an applied differential pressure. The information was gathered using the Motor-Operated Valve Analysis and Test System (MOVATS).

NRC IEB 85-03 requires the High Pressure Cooling Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems' Motor-Operated Valves (MOVs) to be reviewed and verified operable at any differential pressure the valve could be subjected to.

IEB 85-03 requires as much correlation as possible between calculated thrust and the measured thrust needed during valve stroke under differential pressure. Not all MOVs can be stroked with an appropriate differential pressure. The differential pressure seen during accident or inadvertent valve operation may be much higher than can be achieved in a controlled test where prevention of system/component damage is mandatory.

Several valves were chosen for evaluation based on accessibility and available differential pressure. V23-14 was one of the MOVs to be tested. V23-14 was tested during plant operation since it is a steam isolation valve and cannot be tested during shutdown. It also has an acceptable differential pressure near its accident pressure, and is a gate valve.

SAFETY EVALUATION SUMMARY:

The HPCI System mechanical equipment and piping are Safety Class 2. The HPCI System electrical control equipment is safety class electrical.

The HPCI System provides make-up water to the reactor vessel during accident conditions, when sufficient vessel pressure exists, in order to prevent release of radioactive materials to the environment as a result of inadequate core cooling. The HPCI System is a Core Cooling System as described in FSAR Section 6.

During the periods in this special test procedure when HPCI-14 was de-energized to hook up and remove the MOVATS equipment, the HPCI System could not perform its automatically safety function. During these periods, the RCIC System was relied on to maintain sufficient coolant in the reactor vessel for any "abnormal operational transient" and ADS/LPCI was available for DBA conditions.

Prior to starting this special test, the RCIC System was demonstrated to be operable. This provided assurance that RCIC was available during the periods when the HPCI-14 valve was de-energized and the HPCI System could not automatically function. During the actual testing, when the HPCI-14 valve was energized, the HPCI System would automatically operate just as it would during any other operability.

During the periods when the HPCI-14 valve was de-energized, the HPCI System could not operate automatically, but HPCI-14 could have been returned to service quickly to allow the HPCI System to provide a make-up water for "DBA conditions".

This test did not present significant hazards not described or implicit in the Safety Analysis Report, and there is reasonable assurance that the health and safety of the public was not endangered.

EJT/mes:ES0014.1/EJT