Indian Point 3
Nuclear Power Plant
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New York Power Authority

Joseph E. Russell
Resident Manager

December 24, 1991
IP3-NRC-91-034

Docket No. 50-286
License No. DPR-64

Mr. Thomas T. Martin, Regional Administrator
Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Dear Mr. Martin:

Subject: **Code of Federal Regulations 10CFR50.59**
Changes, Tests and Experiments

This letter and Attachment I constitute the 1990 Annual Report for the period January 23, 1990 to January 22, 1991 for changes, tests and experiments for the Indian Point #3 Nuclear Power Plant as required by 10CFR50.59.

The Authority has reviewed each change, test or experiment to ensure that the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased; the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created; and the margin of safety as defined in the basis for any technical specification has not been reduced. It was concluded that these changes, tests and experiments do not constitute any unreviewed safety questions.

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Should you or any of your staff have questions concerning this matter, please contact Mr. Michael Peckham, General Manager Support Services at (914) 736-8041.

Sincerely,

Joseph E. Russell
Resident Manager
Indian Point #3 Nuclear Power Plant

JER:mls

Enclosure: Attachment I

cc: U.S. Nuclear Regulatory Commission
Documents Control Desk
Mail Station PI-137
Washington, D.C. 20555

IP3 Resident Inspector's Office

MOD 86-03-138 IS REV. 1	*INTAKE STRUCTURE ENCLOSURE*

Description and Purpose

The Intake Structure Enclosure project, Phase II, provided a building to protect the circulating water pumps, service water pumps and traveling screens from severe weather conditions. It is a single story structural steel framed building covering the Intake Structure floor area at 15 foot elevation, west of the plant and adjacent to the Hudson River. This modification has been partially completed.

Summary of Safety Evaluation

The existing concrete structure has been evaluated for the additional loading imposed by the enclosure and found to be structurally adequate. The entire Intake Structure Enclosure was designed to withstand seismic loads due to an earthquake equivalent to the plant Safe Shutdown Earthquake (SSE). Additionally, the framed grating enclosure around the vital area of the service water pumps is designed to withstand impact loading due to tornado generated missiles. To facilitate installation of the enclosure, the service water pump screen bypass gate control cabinets had to be relocated. This in no way compromised the reliability of the bypass gate valve system.

MOD 87-03-019 IS REV. 1	*RISTROPH TRAVELING WATER SCREENS*

Description and Purpose

Eight modified Ristroph screens, two service water and six circulating water, were installed in place of the eight conventional traveling water screens at the Indian Point #3 Intake Structure. The new screens have been outfitted to collect and return fish and debris to the river. The new screen system is composed of three major elements: the screen assemblies, the screen wash system and the flume system/river return system. The new system reduces fish mortality in the Hudson River at the Indian Point Unit #3 Plant. The screens were designed for continuous operation. This will result in a cleaner screen and an improved flow path. This modification has been partially completed.

Summary of Safety Evaluation

The new service water Ristroph traveling screens have been seismically analyzed for structural integrity, but not for operability, since they are not safety related. The new screens will have no effect on the supply of water to the safety related service water pumps. Failure of any portion of this modification will have no adverse impact on the safety related service water pumps. The loadings imposed on the safety related intake structure by the Ristroph modification were analyzed and determined to be acceptable. The new screens are designed for continuous operation. This results in a cleaner screen and an improved flow path.

MOD 87-03-144 SIS REV. 0	*REPLACEMENT OF FLOWMETERS FI-929 AND FI-983*

Description and Purpose

This modification replaced flowmeters FI-929 and FI-933 and installed a new flowmeter FI-929B. FI-929 was relabeled as FI-929A. The flowmeters are utilized in the performance testing of the safety injection and Vapor Containment spray functional tests. The new flowmeters FI-929A and FI-983 replaced discontinued models. The additional flowmeter, FI-929B, was installed in a bypass line with isolation valves around FI-929A in order to enhance the detection of fluid flow during the Residual Heat Removal (RHR) and Low Head Injection Line check valve leakage tests. FI-929B has a lower flow rate range than FI-929A.

Summary of Safety Evaluation

These flowmeters are only used for performance testing and are normally isolated by valves that are closed and locked. Replacing the flowmeters FI-929 and FI-933 and adding an in-line flowmeter, FI-929B, did not affect any safety analysis in the FSAR. The flowmeters, valves and additional piping were seismically analyzed and installed in accordance with QA Category I procedures to preclude the release of radioactivity due to a system failure. Since the flowmeters are only used to test equipment performance, their malfunction would not impact the FSAR. Moreover, their operation or malfunction in no way reduces the margin of safety as defined in any Technical Specification. This modification did not involve a change to Indian Point #3's Technical Specifications.

MOD 88-03-035 SIS REV. 1	*HIGH STEAM FLOW SAFETY INJECTION (SI) TIME DELAY*

Description and Purpose

This modification reduces inadvertent Safety Injection System actuations due to instrumentation lags in the Engineered Safeguards System's high steam line flow, low average temperature/low steam line pressure coincidence circuitry. A time delay relay installed in each train of this circuitry compensates for lags in instrumentation by delaying the circuit output, thus reducing inadvertent engineered safeguards SI actuations.

Summary of Safety Evaluation

An analysis was performed that justified the incorporation of a time delay into the output of the high steam flow coincident with low average temperature or low steam pressure circuitry. The analysis has determined that this delay should be no greater than six seconds. This analysis established that for a postulated accident, the Vapor Containment pressure is continuously rising between 0 and 6 seconds and that a maximum six second delay in the Safety Injection actuation signal high steam flow coincident logic will result in a Vapor Containment pressure of 33.61 psig, which is lower than the Technical Specification limit for Vapor Containment pressure (42.42 psig). Therefore, the applicable acceptance criteria for the Indian Point #3 steamline rupture events are still met. Two AGASTAT Model No. E7014 relays were seismically mounted in the rear compartment of Control Room Racks G4 and G6 and they are nuclear qualified to IEEE standards 323-1974 and 344-1975 and have an operating range of .7 to 7 seconds. The additional weight of the relays, 2.57 pounds each, on each rack was evaluated and found negligible compared to the weight of the rack. The power requirements and contact ratings for these relays are well within the margins of the circuitry. Since this modification resulted in a change to the Technical Specifications, pre-implementation approval by the NRC was required and was granted.

MOD 88-03-050 SG REV. 1	*REPLACEMENT OF BLOWDOWN AND SHELL DRAIN PIPING ADJACENT TO THE STEAM GENERATORS*

Description and Purpose

The original 2 inch carbon steel blowdown piping adjacent to the Steam Generators and common 2 inch blowdown piping from the Steam Generators to the crane wall penetration were replaced with 2 - 1/2 inch and 4 inch stainless steel piping, respectively. Also, the original 1 inch carbon steel shell drain piping from the Steam Generator was replaced with 1 inch stainless steel piping. This modification has been partially completed, with the completion of this modification for the #34 Steam Generator only. This modification still needs to be completed for #31, #32 and #33 Steam Generators.

Summary of Safety Evaluation

The larger piping is capable of handling intermittent increased blowdown flowrates without flashing occurring in the piping. The material of construction was changed to stainless steel for improved erosion resistance. All new piping and fittings are designed, fabricated and tested in accordance with the requirements of ANSI B31.1, "Power Piping" and are classified as seismic class I and QA Category I.

All shop butt welds were 100% radiographed, field butt welds were 10% randomly radiographed, and all remaining welds were liquid penetrant tested. This testing program went beyond the code test requirements. All new piping was designed for 600°F and 1085 psig which is consistent with the shell side design, temperature and pressure of the new Steam Generators. Stress analysis results show that the calculated stresses are below the maximum allowable stresses for the various plant operating conditions.

NSE 88-03-135 CCW REV. 1	*REVISED SETPOINT FOR COMPONENT COOLING WATER (CCW) HEAT EXCHANGER OUTLET TEMPERATURE ALARM*

Description and Purpose

The purpose of this evaluation was to determine the effect of raising the alarm setpoint for the CCW heat exchanger temperature alarm from 105°F to a value of 110°F.

Summary of Safety Evaluation

This change was implemented to assure the operability of the component cooling water loop of the Auxiliary Coolant System by alerting the Operators that the discharge temperatures of the component cooling water heat exchangers are approaching the design temperature limit of various components utilizing cooling water. The establishment of a setpoint of 110°F, in lieu of the design limit of 120°F ensures this.

MOD 89-03-074 RCS REV. 0	*REACTOR COOLANT SYSTEM (RCS) LEVEL INDICATION FOR MID-LOOP OPERATION*

Description and Purpose

This modification installed a mid-loop level monitoring system consisting of two parallel level columns, a video camera, dedicated lighting located in the Vapor Containment, and a video monitor located in the Control Room to enhance monitoring Residual Heat Removal (RHR) operation at reduced RCS inventory. This system provides the first line of protection against an RCS mid-loop loss of decay heat removal event by providing indication of RCS mid-loop level to the Operators. The maintenance of RCS mid-loop level will preclude RHR pump damage by preventing RHR pump suction vortexing through the preservation of an adequate net positive suction head for the RHR pumps. This modification satisfied a commitment to the NRC in NYPA's response to Generic Letter 88-17.

Summary of Safety Evaluation

This modification serves to upgrade the design of the mid-loop monitoring system and makes no changes to equipment, operating modes or parameters that create new failure modes or plant transients. The design of this modification is consistent with and in accordance with the requirements regarding channelization, separation criteria, seismic criteria and ASME Section XI.

NSE 89-03-084 EDG REV. 0	*480 VOLT AC EMERGENCY DIESEL GENERATORS CONTROL CIRCUIT TIMING RELAYS AND PRESSURE SWITCHES' SETPOINTS AND TOLERANCES DETERMINATION*

Description and Purpose

The purpose of this evaluation was to determine and establish acceptable setpoints and tolerances for the timing relays, and lube oil and jacket water pressure switches associated with the control circuitry of the Emergency Diesel Generators.

Summary of Safety Evaluation

1) The implementation of the new setpoints for Overcrank Timers, OCT_1 and OCT_2 will improve diesel engine reliability. The new setpoints that have been established prevent total depletion of the starting air tanks if the emergency diesel generators do not start on demand.

2) The new time setting for the shutdown SDR relay in conjunction with the new setpoint (45 ± 5 psi) for the lube oil pressure switches permit a more efficient shutdown of the emergency diesel generators' engines.

3) The implementation of the new setpoint (30 ± 3 psi) for the lube oil pressure switch in conjunction with the lube oil pressure timers will prevent inadvertent activation of the 86 lock out device, thereby increasing emergency diesel generator reliability.

4) Tolerance values are based on the specifications supplied by the emergency diesel generator vendor.

It can be affirmed that the new timing relay and pressure switch contact settings will not degrade the designed safety or instrument functions of plant equipment.

NSE 89-03-213 CCW REV. 1	*REQUIRED COMPONENT COOLING WATER (CCW) SETPOINT CHANGES FOR 95° F SERVICE WATER TEMPERATURE*

Description and Purpose

The purpose of this nuclear safety evaluation was to determine the effects of changing the following component cooling water system alarm setpoints.

1) Reactor Coolant Pump (RCP) motor low flow alarms (FIA-613, 616, 619, 622) from 125 to 155 gpm.

2) RCP thermal barrier low flow alarm (FIA-625) from 80 to 100 gpm.

3) RCP thermal barrier high flow isolation (FIA-625) from 150 to 175 gpm.

4) CCW pump header low pressure alarm/automatic start of standby pump (PIC-600A, 600B) from 80 to 100 psig.

5) CCW heat exchanger outlet high temperature alarm (TIA-602A, 602B) from 105°F to 110°F.

6) Spent Fuel Pool (SFP) high temperature alarm (TIC-651) from 125°F to 135°F.

The CCW System is cooled by the service water drawn from the Hudson River. The original design of this system was based on a maximum 85°F river water temperature. Due to 1988 meteorological conditions, analyses were performed to increase the design bases temperature to 95°F. As a result of this, the above listed setpoints required change.

Summary of Safety Evaluation

These changes were implemented to assure the operability of the component cooling water loop of the Auxiliary Coolant System, provide pump runout protection, and by alerting the Operators that the components serviced by component cooling water are approaching design temperature limits.

MOD 89-03-251 RHR REV. 0	*RESIDUAL HEAT REMOVAL (RHR) PUMP MINIFLOW MODIFICATION*

Description and Purpose

This modification was intended to fulfill NYPA's commitment to the NRC regarding IE Bulletin 88-04 to upgrade the RHR System such that the potential for "dead heading" a RHR pump is eliminated. The modification consisted of adding new 3 inch minimum recirculation lines for each RHR pump. Each new recirculation line is sized to pass the manufacturer's recommended minimum flow (300 gpm per pump) and is provided with a manual throttling valve to reduce pump discharge pressure. Permanent flow measurement instrumentation was installed in each new recirculation line for initially setting the position of throttling valves, AC-841 and 842, and for monthly pump functional testing. This modification has been partially completed with the manual valve handwheel reach rods and insulation remaining to be installed.

Summary of Safety Evaluation

This modification fulfills NYPA's commitment with regard to NRC IE Bulletin 88-04 by providing individual recirculation lines for each RHR pump. The addition of individual recirculation lines effectively segregates the RHR pump discharge lines and prevents the strong-weak pump interaction phenomenon with which the NRC IE Bulletin is concerned. This modification did not reduce the margin of safety as defined in the basis for any Technical Specification.

MOD 89-03-253 FHS REV. 0	*MAXIMUM DENSITY RE-RACKING*

Description and Purpose

This modification replaced the twelve (12) spent fuel storage racks with 12 new maximum density racks. This increased the spent fuel storage in the pool from 846 locations to 1345 locations. A number of storage locations will be utilized to store fuel assemblies with damaged upper nozzles.

Summary of Safety Evaluation

During the design of the new maximum density racks and evaluation of the pool structure and cooling ventilation system, a large number of reports were generated, reviewed, and approved. Based on the information contained in these reports and analyses, and with pre-implementation approval by the NRC, there is reasonable assurance that the health and safety of the public was not endangered by operation when the re-rack was being accomplished. Modification installation activities were conducted in compliance with the Authority's procedure within the NRC's regulations. Thus, this project was not inimical to security or the health and safety of the public. Since this modification resulted in a change to the Technical Specifications, pre-implementation approval by the NRC was required and was granted.

MOD 90-03-004 WDS-S REV. 0	*REPLACEMENT OF SPENT RESIN DISCHARGE PIPING*

Description and Purpose

This modification eliminated "crud" traps in the spent resin discharge line and improved the flushing capabilities of this system in order to achieve lower area radiation dose rates. The unused spent resin drum fill manifold was removed and replaced with a continuous pipe. The previously existing drum fill valve manifold was also removed for better space utilization. The electrical wiring for power, control, indication and heat tracing associated with these manifolds were also removed.

Summary of Safety Evaluation

The new piping improved the flushing ability to remove any resin left in the pipe following a transfer. All piping involved with this modification is Seismic Class III, QA Non-Category I which is non-safety related. This modification does not involve nor is it located near any equipment which is safety related.

MOD 90-03-011 RCS REV. 0	*MID-LOOP ULTRASONIC LEVEL MEASURING SYSTEM*

Description and Purpose

This modification installed a Westinghouse Ultrasonic Level Measuring System (ULMS). The ULMS provides indication of water level in the reactor coolant piping during nonpower operations in an effort to mitigate the possibility of loss of decay heat removal as a result of air ingestion by the Residual Heat Removal pumps during mid-loop operation. New cables were also installed as part of this modification. Some cables were installed in trays and the rest were run in newly installed conduit. The maximum current draw of the system during operation is less than 1 amp and does not overload the panels providing power. This modification has been partially completed with further testing required.

Summary of Safety Evaluation

This modification met the intent of NRC Generic Letter 88-17. The ULMS provides indication only to the Control Room Operators. The system provides no safety related function. All components were seismically mounted to prevent any systems interaction occurrence.

RECLAS 90-03-021 MS	*RECLASSIFICATION OF THE MAIN STEAM NON-RETURN CHECK VALVES MS-2 VENT VALVES (MS-57)*

Description and Purpose

The purpose of this reclassification was to reclassify the vent valves (MS-57) and their associated piping for each of the four Main Steam non-return check valves, MS-2, from QA Category I to QA Non-Category I.

Summary of Safety Evaluation

The vent valves for the MS-2 valves do not constitute a vapor containment isolation, nor are they relied upon to limit the release of radioactivity. While the MS-2 valves themselves perform a safety related function, the vent valves and their associated piping are located downstream of the MS-2 valves seating surface and have no impact on their ability to perform their safety function. The MS-57 valves do not have a safety related function; their purpose is to vent downstream of the MS-2. Failure of any of the vent valves and their associated piping would not effect the structural rigidity of the MS-2 valves, and would have no impact on

the seismic classification of Class I on the Main Steam piping, which allows reclassification of the vent valves and their associated piping to QA Non-Category I.

RECLAS 90-03-074 VS/SGME REV. 0	*RECLASSIFICATION OF STEAM GENERATOR MAINTENANCE EXHAUST SYSTEM (SGMES)*

Description and Purpose

This reclassification downgrades the Steam Generator Maintenance Exhaust System (SGMES) from QA Category I to QA Category I with Non-Category I boundaries. This included both the duct work as well as the fans used to exhaust air from the Steam Generators during periods of maintenance.

Summary of Safety Evaluation

The reason for the system's original QA category classification has been disproven as per the design documents behind its initial installation. The possibility of seismic interaction was addressed only to the point that the installation of any platforms associated with the system was to be Category "A", the nuclear grade designation of Con Edison, the original owners of Indian Point #3. Therefore, the ventilation duct work and fans associated with the system shall be restrained by supports considered to be nuclear safety related. The SGMES was reclassified to having QA Category I supports with Non-Category I components, to satisfy system interaction concerns and its functional state was revised from AC "active" to PC "passive."

NSE 90-03-119 RCP REV. 0	*REACTOR COOLANT PUMP SEAL RETURN FLOW TEST EVALUATION*

Description and Purpose

The purpose of this safety evaluation was to evaluate that no unreviewed safety questions would be created during the planned performance of tests ENG-403A, ENG-403B, ENG-403C and ENG-403D titled, "RCP #1 Seal Leakoff Flow Verification", when the reactor is critical.

Although the performance of these tests did not involve the operation of any equipment in a manner outside its design function, this safety evaluation was prepared to address hypothetical events and their potential consequences.

Summary of Safety Evaluation

Performance of the tests in no way interfered with the continued safe operation of the plant. The test procedures themselves did not involve the operation of any equipment in any configuration other than its intended design. The test procedures were prepared to reduce the possibility of human error.

It is concluded that potential failures would only be caused by operator error and not as a result of the test. Furthermore, any failure would result in an economic concern and not any unanalyzed accident or event. The hypothetical events which could occur are bounded by the Indian Point #3 FSAR and are addressed in plant operating procedures.

RECLAS 90-03-123 FHS REV. 1	*RECLASSIFICATION OF THE SPENT FUEL POOL BRIDGE / CRANE*

Description and Purpose

The purpose of this reclassification was to reclassify the spent fuel pool bridge/crane and index system from QA Category I to QA Category M.

Summary of Safety Evaluation

All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a supervisor technically trained in nuclear safety. The geometric configuration of the new maximum density spent fuel racks along with the boron concentration in the spent fuel pool ensure adequate subcritical margins under all postulated conditions. The spent fuel pool bridge and hoist are designed with interlocks so that fuel cannot be raised above a safe shielding depth for the safety of operating personnel. The motions of the spent fuel pool bridge and hoist are limited to low speeds and extreme caution is exercised during fuel handling to prevent the fuel assembly from striking another fuel assembly or structure. In the fuel storage pool, the fuel assemblies are spaced in a pattern which prevents possibility of a criticality accident.

Based on this evaluation, it was concluded that the spent fuel pool bridge/crane is not safety related equipment because the consequences of a fuel handling incident resulting from a failure of the spent fuel pool bridge/crane shall not result in a potential offsite exposure in excess of the 10CFR Part 100 Guidelines. Furthermore, the spent fuel pool bridge/crane is classified as Seismic Class III, it has been designed with seismic considerations such that it will retain its position and structural integrity and will not endanger Seismic Class I structures and equipment during Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) seismic events.

MMP 90-03-129 MULT REV. 0	*REMOVAL OF VAPOR CONTAINMENT (VC) ISOLATION VALVES UH-37 AND UH-38*

Description and Purpose

The purpose of this modification was to remove valves UH-37 and UH-38, to cut and cap lines #20 and #95 at both ends of their respective penetrations S and T, to retire the Isolation Valve Seal Water (IVSW) line to both valves UH-37 and UH-38, and to connect the capped pipe sections through the penetrations to the Weld Channel and Containment Penetration Pressurization System (WCCPPS). This eliminated the maintenance and testing requirements for valves UH-37 and UH-38 and upgraded the structural integrity of VC penetrations S and T while maintaining them as pressurized double barrier piping penetrations between the Vapor Containment atmosphere and the outside environment.

Summary of Safety Evaluation

The removal of valves UH-37 and UH-38 has been evaluated in terms of its effects on Vapor Containment integrity on the operation of the WCCPPS and on the structural integrity of penetrations S and T and lines #20 and #95 and their supports. Vapor containment integrity is maintained because lines No. 20 and No. 95 were capped on both sides of penetrations S and T and the piping in between will be pressurized by the WCCPPS. Since the additional volume of piping to be pressurized by the WCCPPS is insignificant and no WCCPPS make up flow is expected because of the high reliability of a welded joint, the performance of the WCCPPS was not impacted. The loads on penetrations S and T was reduced as a result of removing the valves and capping the lines at the penetrations. Consequently, the structural integrity of the penetrations is improved. It can be concluded that this modification did not present any unreviewed safety questions and did not reduce the safety margin of the plant. Since this modification resulted in a change to the Technical Specifications, pre-implementation approval by the NRC was required and was granted.

MOD 90-03-138 CVCS REV. 1	*CHARGING PUMP DISCHARGE CHECK VALVE REPLACEMENT*

Description and Purpose

This modification replaced the six 3 inch check valves located on the discharge side of the three charging pumps and upgraded the associated piping supports. The original valves had exhibited body to seat cracks and leakage problems. The root cause was determined to be due to flow-induced vibration from the chattering or hammering effect by the valve piston assembly not being completely lifted. By replacing the 3 inch check valves with 1-1/2 inch size check valves and thus reducing the valve seat area, sufficient pressure drop will be maintained to lift the valve piston assembly completely. This modification should reduce the flow-reduced vibration effects on the valves.

Summary of Safety Evaluation

All work associated with this modification was performed in accordance with QA Category I criteria. All materials used were QA Category I and all welds were applied by qualified welders. This ensured the integrity of the modification under all plant operating conditions, including seismic events, and also mitigated any system interaction concerns.

NSE 90-03-180 RCC REV. 0	*CYCLE 8 RELOAD SAFETY EVALUATION AND CORE OPERATING LIMITS REPORT (COLR) EVALUATION*

Description and Purpose

This evaluation documented that the cycle 8 COLR and Reload Safety Evaluation (RSE) provided by Westinghouse Electric Corporation was sufficient to demonstrate that the cycle 8 core will not adversely affect the safety of the plant.

Summary of Safety Evaluation

The cycle 8 RSE was prepared by Westinghouse subsequent to the generation of a fuel assembly loading pattern. Prior to reloading the reactor vessel, this pattern was verified as meeting thermal-hydraulic requirements for full power cycle 8 operation.

MMP 90-03-188 STR REV. 0	*REPLACEMENT OF STEAM AND FEED LINE VAPOR CONTAINMENT (VC) PENETRATIONS' EXPANSION JOINTS*

Description and Purpose

Due to the failure by cracking of the expansion joints of the Main Steam and Boiler Feedwater lines VC penetrations, and the subsequent Weld Channel and Containment Penetration Pressurization System (WCCPPS) leakage, replacement expansion joints were installed at these penetrations. The new expansion joints have functional characteristics identical to those of the originals (meeting requirements of UE&C Spec. 9321-05-248-22) but are of improved material. ASME SB168 Inconel 600 alloy was chosen as the material of construction.

Summary of Safety Evaluation

ASME SB168 Inconel alloy was deemed to be an ideal material, specifically in terms of resisting the chloride-abetted corrosion which was determined to be the most probable cause of initiating the cracking in the bellows. All applicable sections of the ASME Boiler and Pressure Vessel Code were utilized in this modification. Materials, welding processes, welder qualifications, NDE, and overall design are covered by the Code. Authority approved procedures and drawings were the actual implementing documents for the fabrication and installation phases and the task was performed in accordance with the Authority's QA Category I Program.

MMP 90-03-190 RCC REV. 0	*SETPOINT CHANGE TO ROD INSERTION LIMIT ALARMS*

Description and Purpose

The rod insertion limit alarms in the Indian Point #3 Control Room notify the Operators when the control rod position is at or near the greatest allowed insertion for that power. Core design analyses for cycle 8 included a different alarm curve, and the Control Room instrumentation was adjusted to incorporate it. The original rod insertion limit curve allowed insertion of D-bank to 22% of the full power and insertion of C-bank to 48% at zero power. The cycle 8 Core Operating Limits Report (COLR) and Plant Operating Procedure 2.3 (POP-2.3) allowed deeper insertion of control banks at zero power. The rod insertion limit alarms in the Control Room were re-adjusted to reflect these new limits. This was accomplished by resetting the limit computer using existing plant procedures. There are two limit alarms, one alarm when rod

position is approaching the limit, the other when rod position is at the limit. Both these alarms still function in exactly the same manner as they did before the change.

Summary of Safety Evaluation

All accident and safety evaluations relating to this setpoint change were reviewed and approved in advance by Westinghouse. The cycle 8 Reload Evaluation (RSE) and the cycle 8 COLR reflect this change. Specifically, the new rod insertion limits were reviewed for their effects on cycle peaking factors, shutdown margin, rod ejection, trip reactivity and inadvertent dilution. In all cases, the new setpoints were found to be acceptable.

CLAS 90-03-203 FHS REV. 0	*RECLASSIFICATION OF MAJOR COMPONENTS OF FUEL TRANSFER PORTION OF FUEL HANDLING SYSTEM (Except Supports)*

Description and Purpose

The fuel transfer subsystem of the Fuel Handling System (FHS) provides transport for new and spent fuel between the Fuel Storage Building (FSB) and Containment and was previously classified as QA Category I. The purpose of this classification was to reclassify components of this sub-system to Category M. Those bolts, beams and welds that are required to directly support these components and/or attach them to the walls and floor of the refueling cavity or spent fuel pool shall remain QA Category I because of their potential direct impact on safety related structures and components.

Summary of Safety Evaluation

The analyses of the probabilities of failures in the equipment associated with the fuel transfer subsystem have been previously evaluated in the FSAR and demonstrate that offsite exposures in excess of 10CFR100 Guidelines would not occur since the ultimate source of safety associated with an irradiated fuel assembly outside of the reactor core is the borated water surrounding it at all times. The classification of the fuel transfer subsystem did not affect in any way the borated water in the vapor containment's refueling cavity nor the Spent Fuel Pool Building's spent fuel pool.

MOD 90-03-237 FHS REV. 0	*RETRIEVAL OF FUEL ASSEMBLIES SUSPENDED FROM THE REACTOR VESSEL UPPER INTERNALS*

Description and Purpose

This modification focused on the design and implementation of the devices and procedures used to safely retrieve the two fuel assemblies which were inadvertently suspended and attached to the reactor upper internals during the course of the cycle 7/8 refueling outage. The designs of the fuel assembly recovery canisters or baskets, recovery table and manipulator crane hoist and rigging were reviewed in terms of their materials, dimensions, installation concerns, internal stresses, stability and fuel cooling capability. The baskets were functionally tested using mock-ups of the recovery area and fuel assemblies. The procedures that were utilized during the recovery of the fuel assemblies were designed with the following considerations: the location of the recovery baskets, the operation of the trolley and polar crane, the effects of the Residual Heat Removal flow and transfer motion on the internals assemblies package, use of visual aids and communications, and radiological exposure. The final portion of this evaluation dealt with various scenarios that could have had a possible impact upon the safety of the plant.

These included:

- The dropping of one or both of the fuel assemblies into the core.
- The dropping of one or both of the fuel assemblies outside of the core.
- Polar crane brake seizure.
- Vertical control failure on the polar crane.
- Fuel cladding failure before or during the loading of the elements into the fuel container assembly.
- Failure of the uncovered fuel.
- Presray (cavity) seal failure due to a dropped assembly.
- Partial failure of the Presray seal.
- Loss of Vapor Containment integrity.
- Potential failure of the nozzle dam at #34 Steam Generator.
- Loss of off-site power.

Summary of Safety Evaluation

The events considered included those which could be directly caused by the actions of this modification as well as unrelated events which could impact the safety of the plant or further degrade plant conditions if they occurred simultaneously with the recovery of the fuel assemblies. It was determined that the recovery operation could be completed within the bounds of the Indian Point #3 FSAR.

NSE 90-03-273 RV REV. 0	*EVALUATION FOR USING THE REACTOR VESSEL'S INTERNALS AFTER PERFORMING TEMPORARY MODIFICATION #1243*

Description and Purpose

The purpose of this safety evaluation was to evaluate the reinstallation of the reactor vessel's upper internals assembly back into the refueled reactor vessel following the repair and inspection efforts on the reactor vessel's upper internals fuel assembly guide pins. The repairs on the guide pins were performed under temporary modification #1243.

Summary of Safety Evaluation

The reassembly of the reactor during cold shutdown, utilizing the reactor vessel's upper internals which were inspected and repaired did not create any geometry change which could jeopardize decay heat removal. Similarly, the reinstallation of the repaired and inspected reactor vessel's upper internals assembly did not cause any geometry changes that would jeopardize the subcriticality margin of the cycle 8 core. Based on the above, the reactor vessel's upper internals could be returned to the refueled reactor core without causing any damage to the fuel assemblies or other reactor components.

NSE 90-03-280 RHR REV. 0	*EVALUATION OF RESIDUAL HEAT REMOVAL (RHR) PUMP NPSH UNDER LONG-TERM POST-LOCA RECIRCULATION CONDITIONS*

Description and Purpose

The purpose of this evaluation was to analyze the Net Positive Suction Head (NPSH) of the RHR pumps during the long-term, post-LOCA recirculation mode of operation and to identify the operational conditions necessary to ensure that pump cavitation would not occur.

Summary of Safety Evaluation

The Emergency Operating Procedures ES-1.3 and ES-1.5 were revised to provide steps to manually throttle valves, SI-HCV-638 and SI-HCV-640 so that the combined flow through the two (2) RHR heat exchangers does not exceed 4500 gpm when supplied from either RHR pump operating alone in order to preclude cavitating the operating RHR pump. This change limited RHR pump flow such that minimum pump NPSH requirements are always met with margin.

This procedural change ensured that long-term, post-LOCA recirculation will be accomplished with at least minimum safeguards equipment available. The change did not have any effect on structures important to safety. Even though the structures associated with the RHR pumps are designed to withstand seismic as well as operational loads, none of these were altered as a result of the change. Other systems and components will continue to be operated as before.

With the minimum safeguards equipment available to mitigate the consequences of the long-term, post-LOCA recirculation phase, no margin of safety as defined in the bases for Technical Specification was reduced.

MMP 90-03-299 WCCPP REV. 0	*ELIMINATION OF VAPOR CONTAINMENT (VC) AIRLOCK SOLENOID VALVES AND ASSOCIATED EQUIPMENT*

Description and Purpose

The Weld Channel and Containment Penetration Pressurization System (WCCPPS) solenoid valves and related pressure indicating gauges located at the personnel access airlocks into the Vapor Containment Building at the 80' and 95' elevations were removed. The removal of this equipment was performed due to the frequent failures of this equipment, the significant effort devoted to this maintenance of this equipment and the analyses which indicated that this equipment was not required to assure VC integrity.

Summary of Safety Evaluation

This modification improved the leak integrity of the WCCPS at the VC airlocks by replacing the solenoid valves and associated components with welded fittings (such as caps, plugs, and piping spool pieces). These fittings will permit the airlock system to be continuously maintained in the safe response mode; no component function (at the airlocks themselves) will have to be assumed to place the system in a required state.

The work was performed to QA Category I criteria to ensure that proper controls were in effect. Testing and inspecting was conducted to verify the adequacy of the original components removal and the installation of new parts.

No new unanalyzed accidents or malfunctions were potentially created, nor was any Technical Specification (or basis) revised by this modification. Therefore, it is concluded that no unreviewed safety question was introduced by this modification.

MMP 90-03-307 EDG REV. 0	*EMERGENCY DIESEL GENERATOR (EDG) FUEL OIL DRIP TANK DRAIN PIPING TO EDG FUEL OIL DAY TANK*

Description and Purpose

This modification was performed to make temporary modification, Jumper #974, a permanent design change. This modification disconnected and capped the interconnecting piping between the EDG fuel oil day tank and the EDG fuel oil drip tank pump discharge. Its purpose was to prevent water and other possible contaminants that may collect in the EDG fuel oil drip tank from being pumped into the EDG fuel oil day tank. This modification was completed on all three EDG's.

Summary of Safety Evaluation

This modification improved plant operability by providing greater assurance that the EDG fuel oil supply system will remain free of contaminants. The EDG fuel oil drip tank can still be drained manually to a separate container instead of the EDG fuel oil day tank; therefore, no new accident scenarios were created. No changes to Technical Specifications were required nor was the margin of safety as defined in the basis for any Technical Specification reduced by the modification.

NSE 90-03-321 NIS REV. 0	*EVALUATION OF COMPLETION OF CONTROL ROD DROP TIMING TEST PARTIAL COMPLIANCE WITH PRE-REQUISITE STEP 2.16 OF 3PT-R4 AND STEP 2.12 OF 3PC-R6*

Description and Purpose

This evaluation reviewed the controls established in the periodic surveillance test 3PT-R4 "Full Length Rod Time Test" to determine if there was an unreviewed safety question in performing the test without full compliance with step 2.16 which requires that all Reactor Protection System Circuitry is operable with all required surveillance tests successfully completed on schedule. This evaluation also addressed the acceptability of performing Rod Position Indication calibrations without delta flux indication or f(△I) penalty.

Summary of Safety Evaluation

The unavailability of delta-flux meters and the f(△ I) penalty in the overtemperature delta temperature trip did not introduce a safety concern during the performance of the Full Length Rod Drop Timing Test and the Rod Position Indication Calibration Test (3PT-R4 and 3PC-R6) because more than sufficient procedural control and protective instrumentation was available and operable. Because power levels up to 90 percent of rated power are allowed by Technical Specification 3.11-B without the excore axial off-set detection system being calibrated, no restrictions should be placed on performing the rod position indication calibration and rod drop timing test because of an inability to calibrate the delta flux meters or the f(△I) penalty. Westinghouse was contacted on these issues and their evaluation documented in a December 17, 1990 transmittal between L. R. Grobmyer and G. L. Fidler support the findings and conclusions of this evaluation.

The f(△I) penalty function for overtemperature delta temperature is not used in any of the FSAR accidents that begin with the reactor in the subcritical condition. This testing did not create the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR since no plant systems or components were modified nor were any new operations conducted. Furthermore, this testing did not reduce the margin of safety as defined in the basis for any Technical specification nor did it require a change in the Technical Specifications since they do not require the f(△ I) penalty as part of overtemperature delta temperature or the delta-flux meters to be operable at hot shutdown.

Performing the control rod drop time test, 3PT-R4 and the rod position indication calibration 3PC-R4 without a calibrated f(△ I) penalty function or delta-flux indication does not compromise plant safety.

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