CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT 2

CHANGES, TESTS AND EXPERIMENTS - 1986

Preface

Discussed herein are descriptions and safety assessments of changes performed at Indian Point Unit 2 completed in 1986. These have been evaluated and determined to meet the following criteria as established by 10 CFR 50.59. It has therefore been concluded that none of these changes represents an unreviewed safety question.

Criteria

1. The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

3. The margin of safety as defined in the basis for any technical specification has not been reduced.

TABLE OF CONTENTS

ITEM	TITLE	PAGE

1. Modification of reload startup physics test program.

During the initial return to power following a refueling shutdown, a series of core physics tests are performed. The objectives of these tests are: (1) to demonstrate that the core performance during reactor operation will not exceed safety analysis and Technical Specification limits, (2) to verify the nuclear design calculations, and (3) to provide the bases for the calibration of reactor insrumentation.

The changes instituted by this modification involve the Hot Zero Power (HZP) and beginning of core life condition tests. The changes include the following:

1. Depletion of movable incore detector flux map performed at low power (below 5%) with all the rods withdrawn and replacing it with a flux map at ≤30% power with all the rods (essentially) withdrawn.

2. Addition of control bank B worth measurement with control banks C and D inserted.

3. Addition of control bank A worth measurements with control banks C, D and B inserted.

4. Replacement of the measurement of isothermal temperature coefficients and boron end points undertaken at the "Control Bank D In" state point, by the measurements at the "All Control Banks (A, B, C and D) Inserted" state point.

Replacing the HZP flux map with additional control rod worth measurements of control banks B and A and a flux map at approximately 30% power does not increase the likelihood of an undetected core loading or design anomaly and therefore does not degrade the safe operation of the plant.

2. Cycle 7/8 refueling-Cycle 8 operation.

A report entitled "Reload Safety Evaluation, Indian Point Nuclear Plant, Unit 2, Cycle 8" was prepared by Westinghouse Electric Corporation.

The report presented an evaluation for Cycle 8 which demonstrated that the core reload did not adversely affect the safety of the plant. All incidents analyzed and reported in the FSAR which could potentially be affected by the fuel reload were reviewed for the Cycle 8 design. The results of new analyses were included, and the justification for the applicability of previous results for the remaining analyses was presented.

It was concluded that the Cycle 8 design does not cause previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the assumptions that: (1) Cycle 7 operation is terminated at 12,360 ± 500 MWD/MTU, (2) Cycle 8 burnup is limited to the end-of-full power capability plus 500 MWD/MTU for power coastdown and (3) there is adherence to plant operation limits as given in the Technical Specifications.

Cycle 8 is the first cycle of all LOPAR fuel assemblies. The NRC has approved the core transition and technical specification changes associated with LOPAR fuel core. In addition, Cycle 8 introduces the Westinghouse Wet Annular Burnable Absorber (WABA) rods. WABA rods are described in the NRC-approved WABA evaluation report, WCAP-10021-P-A, Rev. 1, dated October 1983. The report is applicable to Indian Point Unit 2.

Cycle 8 has a low leakage loading pattern similar to that of Cycles 6 and 7.

The only item relating to reload not specifically addressed in the report involves storage and handling of the Region 10 fuel. The Indian Point 2 fuel storage pool design was based upon a fuel enrichment of 3.5 w/o U-235. The Region 10 fuel has split feed assemblies of nominal enrichments of 3.44 w/o and 3.20 w/o U-235 and does not therefore raise any concerns relating to the Technical Specification limitation of 3.5 w/o. Since the exterior lateral dimensions and configuration of the fuel assemblies are unchanged, they will continue to fit properly into the present fuel racks. Additionally, the nominal fuel assembly total weight for Region 10 is approximately the same as that for Region 9. As a result, the Cycle 7/8 reload does not degrade the fuel handling and storage system's seismic design or normal load bearing capability.

3. <u>Installation of flow restrictor, open vent and flow meter in the 18" service water return line</u>.

This modification involves installation of three items on the 18" service water system (SWS) return line from the containment cooling coils. The first item is a new flow restrictor installed at elevation 43 ft. The orifice restricts flow to 10,000 gpm, thus assuring adequate cooling water from the containment cooling coils during the accident mode of operation. The second item is a new 8" open vent installed at elevation 40 ft. 3 in. It replaced the 3" vacuum breaker. The third item is a new flow meter which was installed at elevation 53 ft. for measuring the total flow from the containment cooling coils. These changes do not impact any accident analyses and in fact increase the ability of the service water system to perform its intended function.

4. <u>Cofferdam for the reactor cavity pit</u>.

This modification involved the construction of a cofferdam in the reactor cavity pit area for the purpose of expediting testing of the reactor cavity it pumps liquid level alarms and actuation. Because the cofferdam purpose is to expedite testing, it serves no safety function.

The cofferdam consists of three panels: side panels A and C and mid panel B. The cofferdam side panels were installed permanently and were restrained so that they could withstand a safe shutdown earthquake (SSE) and not damage surrounding safety equipment. Panel B, a removable structure, is removed after testing is completed and is seismically attached to panel C prior to going above cold shutdown. This ensures an adequate flow path between the area below the reactor vessel and the reactor cavity pit sump for leakage detection.

The cofferdam was placed under the incore instrumentation tubes. Sufficient clearance exists between the cofferdam side panels and the incore instrumentation tubes such that tube thermal expansion does not cause the tubes to touch the cofferdam.

5. <u>Upgrade of intake bay chlorination system for direct use on the service water system</u>.

This modification involved the installation of a new chlorination system at Indian Point Unit 2 to provide a reliable and accurate feed system for injecting sodium hypochlorite into the service water system to prevent micro-organism buildup on fan cooler tubes as well as other components in the nuclear and non-nuclear systems utilizing service water for cooling. Sodium hypochlorite is introduced into the service water intake bay, upstream of the service water pumps, on a continuous year round basis. New feed pumps were provided for this continuous low level chlorine feed, and the existing chlorine feed pumps were retained to be used for periodic high level chlorination.

The following existing equipment was retained and incorporated as part of the new service water chlorination system:

1. The two sodium hypochlorite storage tanks located in the Unit 1 screenwell house.
2. The two centrifugal hypochlorite feed pumps also located in the Unit 1 screenwell house.
3. The electrical components for the feed pumps.
4. The Saran lined steel discharge piping to the two service water chlorine diffusers located in the service water intake bays.

The new equipoment provided is a follows:

1. Two skid mounted positive displacement metering pumps.
2. Flow indicator for existing pumps.
3. Electrical components for new pumps.
4. Saran lined steel piping from existing storage tanks to the new pumps.

5. Saran lined steel piping interconnecting existing service water chlorination piping at the intake bays to the two new vertical rubber lined and rubber covered diffusers. One new diffuser is located between service water pumps #24 and #25, and the second diffuser between service water pumps #22 and #23. The diffusers were inserted through new penetrations the concrete floor. All new above ground saran lined steel piping is heat traced.

These changes do not impact system function or any accident analyses but are intended to provide added assurance of availability/reliability of service water system components.

6. Segregation of battery charger alarms.

This modification was performed to provide the operator with improved status indication of the battery chargers and the DC system. It involved the segregation of the battery ground contract from existing battery charger trouble alarms into four new alarms and the separation of the existing single category "battery trouble" alarm for both chargers #21 and #22 into two annunciator trouble windows.

The new alarm circuitry only performs an indication function and is isolated from the IE circuitry. Therefore, an alarm failure or malfunction cannot degrade the DC system capability to perform its safety function.

The new electrical conduit in the cable spreading room was restrained to withstand a Safe Shutdown Earthquake, so that no damage to surrounding safety equipment can occur.

The cable routing utilized existing penetrations in the central control room floor which was resealed in accordance with the requirements of the plant Fire Protection Program.

7. Interim onsite storage facility.

This modification established an interim onsite facility for the temporary storage of low-level radwaste by modifying elevation 108 ft. of the unit 1 containment building to provide storage for Dry Active Waste only. The process waste will continue to be shipped to existing disposal sites. This is consistant with the provisions of the current version of the proposed amendment to the 1980 LLRW Act.

The modification in the Unit 1 containment consisted of developing elevation 108 ft. to accept approximately three years of Dry Active Waste (DAW).

The scope of work consisted of the following:

1. Provide fire protection and detection consisting of two hose stations and six infrared smoke detectors.
2. Provide radiation detection and air monitoring equipment (Portable).
3. Refurbish the existing heating and ventilation system to prevent freezing of the fire protecion piping.
4. Modify the existing equipment transfer cars to facilitate the movement of radwaste containers from the Chem Systems building into the Unit 1 containment. Modifications included the installation of an 8000 lb. winch, wire rope and blocks.
5. Provide an electrically-powered pallet truck to move DAW boxes and drums to storage locations outside the coverage of the polar crane. Provides als a battery charger for the maintenance of the pallet truck batteries.

The modifications do not impact any Class "A" systems or adversely affect any non class A support systems for Unit 2. The Unit 1 containment is not Seismic I but was designed to withstand postulated seismic and tornado loadings well in excess of any event ever seen in the location of Indian Point.

Flooding is not an event of concern at the Indian Point site.

Floor loadings due to stored containers and portable shielding surrounding them was analyzed and found not to exceed Unit 1 containment design basis floor loadings.

Prior to the operation of the facility, administrative controls in the form of written procedures would cover areas such as fire protection, radiation monitoring, container integrity, and material handling, to permit the storage area to operate with consideratio of the radiological safety concerns in the NRC Generic Letter 81-38.

Based on the above, safe operation of Indian Point 2 is not degraded by the IOS facility.

8. <u>Ruggedized fully integral low pressure rotor program</u>.

The main steam turbine was retrofitted during the 1986 refueling outage with a completely new low pressure (LP) rotor system.

The installation eliminates susceptibility to stress corrosion cracking in the nuclear steam environment, eliminates blade distress and substantially reduces rotor maintenance. The new rotors provide for the long term reliability of the main steam turbine and eliminate the need to conduct the mandatory NRC disc inspections.

A Westinghouse Topical Report, WSTG-4-NP, entitled "Analysis of the Probability of the Generation of Missiles from the Fully Integral Nuclear Low Pressure Rotors" was submitted to the NRC for review in October 1984.

The design of the original turbine auxiliary systems that interface with the low pressure turbines was reviewed by Westinghouse for adequacy and compatibility with the new fully integral low pressure turbine equipment.

The new nonblock LP rotors are machined from a single forging eliminating the need for shrunk-on discs or body welds. The rotors utilize freestanding blades which reduces operating stresses by the elimination of lashing wires and riveted shrouds. Blades are individually replaceable for rapid installation and maintenance. Each LP rotor is fully interchangeable in all positions.

In order to implement the fully integral rotor changeout, modifications were conducted to the inner cylinder #1, inner cylinder #2 and outer cylinder bases for each turbine. A number of stationary components (diaphragms, segmentals, bearings and bladerings, etc.) also required replacement.

Two Duplex Copper-Constantan "ungrounded" thermocouples were installed in each of the six new LP rotor bearings. The thermocouples provide continuous monitoring of the bearing metal temperature.

The safety review performed for the modification included the review of the Westinghouse WSTG-4-NP, which describes the fully integral rotor and differences between it and the previous rotor. The report also assesses the new probability of crack initiation and missile generation and makes comparison with such assessments made for the previous rotor design. The conclusion of this review is that the missile generation hazard is reduced from the previous design and that the safety of the plant is enhanced by this modification.

9. <u>Central control room HVAC upgrade</u>.

The central control room (CCR) air conditioning system was upgraded to improve reliability and provide the environment which promotes plant monitoring and control reliability.

The air conditioning systems for the CCR consisted of a water cooled 25 ton water chiller and chilled water coil in the air plenum for Unit 1, and a 23 ton water cooled direct expansion air conditioning unit with a 7.5 HP supply air fan motor for Unit 2.

The following modifications were performed:

Unit 1

1. The existing chiller, water cooled condenser, chilled water coil, piping and associated controls located in the mechanical room at el. 72 ft. were removed.
2. A 30 ton air cooled direct expansion A/C condensing unit was installed at el. 88 ft. above the mechanical room.
3. The existing chilled water coil was replaced with a direct expansion air-cooled Freon coil.
4. The existing air handling equipment plenum was refurbished and the condensate pan was replaced with a stainless steel pan.

Unit 2

1. The existing A/C unit and associated piping and controls were removed.
2. A 25 ton air cooled direct expansion A/C unit with a 10 HP fan motor was installed. The air handling unit was located in the same location as the existing unit.
3. The ductwork at elevation 72 ft. was modified to accommodate the new air handling units and the air supply diffusers were replaced to improve air distribution.
4. The air handling unit fan motor will get its emergency power supply from the Unit #2 diesel.
5. The existing 5 HP electric motor of the standby fan at elevation 88 ft. 55 in. was replaced with a 7.5 HP to increase the esternal static pressure of the fan.

Replacement equipment with associated piping and ductwork modifications which are part of the boundary between the CCR and the outside environment are capable of withstanding an SSE and still maintain that boundary. The Unit 2 A/C fan and motor and the spares are also capable of withstanding an SSE and still be functional. Equipment and associated piping which is not part of the CCR boundary in the mechanical room was seismically restrained to insure that the CCR boundary and ventilation system recirculation and air filtration capability will be maintained following an SSE. The remaining Unit 2 service water piping no longer being used was capped and welded and will continue to be capable of withstanding an SSE so that there will be no flow diversion losses from the service water system. Therefore, no accident analysis is impacted or any previously unanalyzed accident created which could degrade safe operation of the plant.

10. Radiation monitoring betterment program.

The purpose of the radiation monitoring system is to provide the operator with information regarding process radiation levels and in certain channels provide control functions that lessen the possibility of release of radiation.

The radiation monitoring betterment program consists of replacing the existing channels with instrumentation wthat will make available to the operator more data with a greater degree of reliability and improved sensitivity.

The new system which is planned to replace the original process reaiation monitoring system is being installed in two phases. Phase 1 installation was completed. The original system will be removed after installation and testing of the new system is complete. The new system is described below as it exists at the completion of Phase 1.

The process radiation monitoring sytem is a digital system with the following major components: individual radiation monitoring units located near the monitored process line; a minicomputer unit located in the technical support center; a CRT display and printer located in the control room; and annunicators located in the control room and the chemical systems control room.

The minicomputer unit includes a console with CRT and typer, magnetic tape drive, and disc drive. It communicates digitally with the individual radiation monitoring units, and processes, records, and displays data.

Each monitor unit monitors a sample of the process fluid which is piped through a bypass loop. The sample is cooled if required. To facilitate maintenance and calibartion, the bypass loop can be isolated and purged.

The liquid monitors utilize sodium iodide gamma scintillation detectors, while the gaseous monitor uses a beta scintillator. Each monitor has a microprocessor which communicates with the minicomputer.

Each monitor will activate an annunication alarm in the event of failure, high radiation, or high temperature.

The minicomputer and the CRT/printer unit are powered from battery-backed inverter. Several monitor units receive power from motor control centers MCC-27 and MCC-26BB which are powered by an emergency diesel generator in the event of loss of other power sources.

The installation of these monitors is such that the safety of the plant is not degraded and that they provide a higher degree of reliability than the monitors they replace.

Information on specific monitors is given below:

<u>Service water from component cooling heat exchangers monitors</u>

Monitors R39 and R40 are powered from MCC-27. They are wired to a control room annunicator, independent of their communications loop through the minicomputer.

Service water return from containment fan cooler units

Redundant monitors R46 and R53 monitor the service water return from all containment cooler units. Each of the channels is hard-wired to a chart recorder in the control room and also to a control room annunciator. They receive power from MC-27. Monitors and piping are designed to withstand an SSE.

Component cooling monitor

Monitor R47 checks the component cooling loop for radioactivity. The activity is recorded on a control room recorder, and high activity initiates a control room annunicator and closes RCV-017. It is powered from MCC-27 and is designed to withstand an SSE.

Waste disposal liquid effluent tank discharge line

This channel, R48, monitors all waste disposal system liquid releases from the plant. Automatic valve closure is initiated by this monitor to prevent further release after a high radiation level is detected. The monitor is hard-wired to a control room chart recorder. It receives power from MCC-27.

Steam generator blowdown

The liquid blowdown from the secondary side of the steam generators is monitored by R49. Upon indication of high activity, an interlock from monitor R49 closes all steam generator contaiment isolation valves. Monitor R49 recives power through MCC-26BB and is designed to operate after an SSE. It will annunciate in the control room independent of its communication loop through the minicomputer. It is hardwired to a chart recorder in the control room.

Waste gas decay tank

This monitor, R50, indicates activity in the waste gas decay tanks. It is hardwired to a chart recorder in the control room and also annunicates in the control room independent of the communication through the minicomputer. It receives power from MCC-27.

Secondary boiler blowdown purification

This monitor, R51, indicates activity in the system effluent. It enables the operator to take corrective action in the event of high activity. It is powered from a Unit 1 motor control center. It alarms in the chemical systems control room independent of its communication loop through the minicomputer.

Liquid waste distillate

This monitor, R54, is powered from a Unit 1 motor control center. It alarms in the chemical systems control room independent of its communication through the minicomputer.

House service boilers

Monitor R59 is powered from a Unit 1 motor control center. It indicates any activity that may be present in the condensate return. It alarms in the control room.

11. Elimination of turbine runback initiation from the overtemperature and overpower delta T circuits.

Spurious turbine runbacks from the overtemperature (OT) and overpower (OP) delta T circuits were a problem at Indian Point 2. Such runbacks caused undesirable transients on the plant. Under certain circumstances, the spurious runback could result in a plant trip which would impose an additional undesirable transient. For this reason the runback initiation from the OT and OP delta T circuits was eliminated.

This was accomplished by disconnecting the load limit runback contacts #1 and #2 initiated by relay RSC-1, initiating relays TCD (OP delta T) and TCD (OT delta T), relays RSC-1 and STD coils located in rack G-4 in the central control room.

The modification involved no unreviewed safety question since no credit has been taken for delta T runback in the safety analysis described in Chapter 14 of the FSAR.

12. Raising the setpoint for direct reactor trip on turbine trip above 10% power.

The 10% power setpoint for a direct reactor trip on a turbine trip was raised to less than or equal to 35% power, using the P-8 permissive. Technical Specifications permit a setpoint of up to 35% power; however, the actual plant setpoint will be approximately 20% power.

The change was made to assure that the reactor does not trip directly on turbine trip during start-up at low power levels, thereby reducing the number of reactor trips during start-ups and challenges to the safeguard systems. The safety analysis supporting the raising of this setpint was approved by NRC in a license amendment no. 107.

The P-8 permissive consists of three-out-of-four power range signals, generated by NC-304 bistables and through series of intermediate relays drive P-8 output relays. The spare contacts of the P-8 relays were connected in parallel with existing turbine auto-stop contacts, which operate RT-15 and RT-16 "Reactor Trip" relays. The P-8 permissive which was used to bypass one-out-of-four RCP trip and one-out-of-four RCS loop flow, was set prior to the change at 50% power. This setpoint was lowered to a new value of less than or equal to 35% power.

13. Removal of boron injection tank, Phase I.

The boron injection tank has been retired from service at Indian Point Unit 2 since 1985. The physical removal of the tank is implemented in two phases. Phase I, which was completed, provides for the electrical disconnection of the associated valves, instrumentation and heat trace and the mechanical disconnection, flushing and depressurizing the tank itself. The complete removal of the boron injection tank will be done under Phase II.

The mechanical portion of the disconnection included the removal of portions of line #199 above the floor at the primary auxiliary building at elevation 80 ft. from the safety injection system. To maintain the seismic integrity of the balance of the safety injection system, a seismic restraint was installed on line #199 near the disconnection point. The cutting and capping of the line #199 suction piping assures that suction to the safety injection pumps from the refueling water storage tank is maintained.

The Nitrogen system associated with the boron injection tank and its connection to the rest of the Nitrogen system was isolated. Thus, the balance of the Nitrogen system will continue to perform its intended function.

Other minor mechanical disconnections were also performed (e.g., CVCS fill line).

The electric portion of the project included the disconnection of power, control, indication and alarm circuits and associated boron injection tank instrumentation, heat tracing and valves. The project was carried out under Class "A" and, where applicable, Class IE procedures.

Accordingly, the physical removal process has been such that remaining systems were not adversely affected and will continue to perform their intended functions.

14. Raising the load cutoff for turbine runback from 70% to 86%.

The load cutoff setpoint for turbine runback was increased to 86%.

The Indian Point 2 turbine runback load reduction was 70% of full turbine load. Based on a safety reanalysis of the rod drop accident performed by Westinghouse, it was determined that the load reduction from 100% to 70% was larger than required and that the setpoint could be increased to 86%. With the raised setpoint, should a turbine runback occur, the transient imposed on the plant would be less severe. In addition, the increased setpoint would result in less economic penalty due to reduced power for real and spurious runbacks and in reduced potential for such a transient leading to a reactor trip.

15. <u>Retire feedwater heaters bypass valve FCV-1150 in place</u>.

The boiler feed pump suction header receives condensate from the normal condensate pump discharge path through the low pressure heaters. A bypass around the low pressure heaters is comprised of the automatic control valve FCV-1150 and a 14 inch line feeding directly from the condensate pumps discharge into the boiler feed pump suction header. The purpose of bypassing the feedwater heaters was to reduce low feedwater addition transients due to postulated load rejection which could challenge the reactor protection system.

Operating experience has demonstrated that the anticipated conditions for this function do not occur thereby obviating its necessity. Furthermore, defeat of this function could eliminate the initiating event for a "reduction in feedwater enthalpy" incident with the worst consequence. Accordingly, no safety or accident analyses could be adversely affected by this change.

Accordingly, the automatic function for FCV-1150 was disconnected and the valve isolated by closing inlet and outlet gate valves CD-4 along with related service valves CD-41 to CD-44.

16. <u>Instrument air system modification</u>.

This modification included: (1) the installation of bypass piping and the removal of the existing refrigerant air dryers, desiccant dryers and filters of the instrument air system, (2) the installation of two new heatless reactivated air dryer/filter sets, and (3) the replacement of a pressure switch.

The modification is part of the station's compressed air system improvement program to provide adequate and reliable instrument and station air supplies. (See also items 17 and 18 of this report.)

The first part of the modification involved the installation of bypass piping and removal of the existing equipment. The bypass piping enabled the removal of the existing equipment without affecting the instrument air system. The removal of the new dryers and filters so that the quality of the air going to the instrument air system was maintained. There are two sets of dryers and filters in parallel. As one set was removed and replaced, the other set remained operational until the first replacement was complete.

The same was done for the second set. The lines that were attached to the equipment being removed but not replaced were capped off to ensure integrity of the instrument air system. These lines were also restrained to prevent pipe whip during a seismic event after the removal of the equipment.

The second part involved the installation of the two heatless reactivated air filter sets. The new set treat the air coming from the air reciver to the instrument air system, are seismically qualified and will improve the quality and reliability of the air supplied by the system.

The third part of the modification involved the replacement of pressure switch 1344 with a newer model for low pressure alarm at the outlet header. The function of the switch remained unchanged.

These changes were performed such that the integrity of the system is maintained and safe operation not degraded.

17. Installation of air purification skid in 3" station air backup to control air system.

As part of the improvement to the instrument air system, an air purification system, including two coalescing prefilters, a heatless desiccant air dryer, and two after filters were installed in the 3" emergency station air backup to the instrument air system.

The air purification skid was connected upstream the existing station air purification equipment. It is located next to motor control center MCC-24 outside the 480-volt switchgear room. The skid and associated piping were seismically restrained.

The modification also involved the relocation of valve IA-30 downstream but within the Indian Point 2 turbine building. By relocating the valve, passage of the air through the air purification skid can be insured when valve IA-30 is closed. In the previous location, upstream the purification skid if the valve was closed no air flowed, if the valve was open the flow would not go only to the skid.

These changes improved the reliability of the backup air supply and did not affect the safe operation of the instrument air system or any other systems.

18. Install direct inter-tie between Unit 1 station air and Unit 2 instrument air purification skids.

A direct connection of approximately 80 ft. of 3" carbon steel pipe and two isolation valves were installed from Unit 1 station air header to the Unit 2 instrument air purification skids #21 and #22. The inactive valves and bypass piping in the instrument air system were removed. The after filter set was removed, and valve PC-1142 was retired and removed.

The installation of a direct inter-tie between the Unit 1 station air and the Unit 2 Instrument air purification skids upstream of the seismic check valve was performed to allow for the work described in items 17 and 18 of this report.

The equipment removed were inactive valves and bypass piping in the instrument air system. Additionally, two existing valves were relocated, one on the new inter-tie and the other on the existing instrument air line upstream of the direct inter-tie. Once valves were removed, welded caps were installed to provide an isolation barrier.

Also as part of this modification was the removal of valve PCV-1142 and the disconnection of the electrical equipment and alarm associated with this valve. The disconnected equipment consisted of a solenoid valve, limit switches, indicating lights and auxiliary relay. The prior operating position of the valve was normally closed. The valve was automatically or manually opened upon loss of instrument air. Since the upgrading of the station air (item 16 of this report), the air quality is the same as that of the instrument air. Thus, the station air system can provide the normal air supply to plant components while the safety-related instrument air compressors are kept in standby, thereby improving their reliability and operational readiness.

The existing manual and check valves upstream and downstream of PCV-1142 provide isolation capability for the instrument air system to protect it from any backflow to the station air system after a seismic event. Check valve IA-20 was relocated adjacent to manual valve IA-21 to provide a smoother flowing piping system. The check valve was provided with a new seismic support to ensure the class boundary.

The after-filter set between valves IA-738 and IS-740 was retired and removed since the new air dryer/filter sets have their own after-filter. The piping between the two valves was capped.

The moisture detector which was already disconnected was removed.

All new piping and piping remaining after the valves have been removed were restrained to prevent pipe whip and remain intact after an SSE. The ability of the instrument air system to perform its function was improved and the performance of any other safety related system was not affected by the modification.

19. <u>Operation and dewatering of CSI mobile demineralization system</u>.

The system, manufactured by Chem Nuclear Systems Inc. (CNSI), is designed to process various aqueous radwaste streams by filtration and/or ion-exchange processes. The system is located in the Unit 1 Chemical Systems Building, a controlled area which has a filtered ventilation system. A CNSI system similar to this has been in use at Indian Point since 1983.

The system consists of the following major components:

1. Demineralization Control Panel
2. Resin transfer skid
3. Stainless steel sluicable demineralizers and headers
4. Disposable pressure vessels
5. Filter vessels with disposable filter cartridges
6. Process vessel shields
7. Interconnecting hoses
8. Booster pump skid

The major advantage of the system is a significant reduction in the waste generation associated with water processing. It was projected that a waste reduction of 20% would be realized when compared to the system previously used.

In evaluating the installation and operation of the system all radiological safety issues were addressed. These issues are outlined in the NRC IE Circular #80-18, dated September 22, 1980. Based on the guidance, a number of administrative and equipment design barriers were enacted and installed to prevent an accident resulting in resin or watter spillage. These controls are described in EH&S procedure 4.200, Rev. 1, and include the following:

Administrative Controls

1. One or more experienced Chem Nuclear Systems technicians will continuously man the demineralizer control panel which is in direct view of all demineralizer equipment while the system is in oepration and are able to stop or start the flow of new resin, spent resin, influent or effluent water with the opening or closing of accessible valves.

2. Before starting daily operation of the system the CNSI technician will verify that:

   a) The RWP/RWA is up to date.
   b) Radiation levels and contamination levels on all equipment are available.
   c) All hoses, piping, and components are free of leaks and are undamaged. All system instruments are operable.
   d) A gamma alarm in the process shield with the highest daily dose is operable.
   e) A leak test (pressure test) is performed any time that a connection is broken per the procedure.

3. By procedue, any plant interface connection that is interrupted must result in CNSI being notified.

4. By referenced CNSI Procedure FO-OP-023 the CNSI technician monitors the High Integrity Container (HIC) liner during its dewatering to assure that its temperature does not increase significantly due to oxidizing agents in the resin. If it does increase, water is immediately added and the demineralizer manager is informed.

Equipment design barriers

1. The system is located in a controlled area with a filtered ventilation system.

2. The sluicable pressure vessels and the disposable pressure vessels are all designed and tested to the applicable ASME code per NRC Regulatory Guide 1.143.

3. All components in the system are hydrotested to 225 psig or greater in order to meet ASME code. Those components that are expected to see operating pressures of 150 psig undergo a 225 psig hydrotest and those expected to see 300 psig undergo a 450 psig hydrotest. In addition, the liquid process sytem and resin transfer hose (stainless steel) are leak tested prior to system operation to operating pressure, and in some cases, above operating pressure.

4. The disposal HIC is equipped with level monitor lights and TV camera to assure that an overflow of spent resin does not ocur. The liner also has a sight glass to observe liner level, and resin flow will also be confirmed by an H.P. who must be present during sluicing. All unnecessary personnel will stand clear of the area.

5. Only one pressure vessel at a time is sluiced so that the maximum amount of resing sent to the 121 ft^3 HIC is 10.5 ft^3.

6. As an additional precaution, the disposal HIC is inside a transportaion cask located within a 10 ft. x 10 ft. area on the Chem Systems Building floor dammed off with an 8 in. high curb. This dam is attached to the floor with caulking and is lined with plastic sheeting with absorbent material placed beneath the plastic sheeting. This dammed area is large enough to hold the entire contents of the HID in the unlikely event of a spill.

7. A HEPA ventilation system is placed at the HIC opening during sluicing operations.

Additional administrative and equipment design barriers are detailed in Procedure 4.200, Rev. 1.

In the highly unlikely event that the above controls and others listed and described in the Procedure fail and resin or effluent water from the waste collection tanks is spilled, the following systems will prevent an uncontrolled release of radioactivity to the environment. A spill inside the Chem Systems Building would be contained by nuclear floor drains inside the building demineralizer cell or the dammed off area around the HIC liner and cask. The dewatering lines in the HIC are filtered to prevent resin from being introduced ito the floor drains. Although the potential for a significant spill is minimal, a conservative analysis assuming a spill of up to 300 gallons of influent water from the Unit 1 waste collection tanks was evaluated. The 300 gallons is based on the assumption of a flow rate of 30 gpm from one evaporator feed pump, for 10 minutes before the waste collection tank valves are shut off.

The offsite dose from this event would be far below the guidelines of 0.5 rem whole body dose and 1.5 rem thyroid dose from gaseous releases as specified in the NRC Circular 80-81. A spill of spent resin would be in a fully hydrated liquid slurry from and will not contain significant amounts of gaseous radionuclides; therefore, the potential airborne release will also be limited. Due to the personnel on hand during system operation any accidental spill would be small and would be quickly detected and aisolated.

There are no seismic considerations related to the use of the demineralizer proces. The equipment does not require any design changes or modifications to existing Chem Systems building equipment and structures.

In order to keep worker exposure ALARA, unnecessary personnel will normally not be present near the Unit 1 demineralizer area during resing transfer operations or near the transfer flexible hose. As per 10 CFR 20.203 (c) (2) the transfer area will be considered a High Radiation Area (> 100 mrem/hr) if the H.P. determines to to be so by procedure.

There are no additional fire protection considerations specific to the process since no additional fire hazard loading is being introduced and no equipment requiring fire protection under 10 CFR 50, Appendix R is involved.

Deviations from Procedure 4.200, Rev. 1, are not allowed without requesting a safety evaluation change of scope review.

Thus no unreviewed safety questions are created by using this system.

20. <u>Containment, reactor cavity and S/G decontamination per EH&S procedure EHS-SQ-4.502,Rev.1</u>.

Procedure EHS-SQ-4.502 Rev. 1., covers the decontamination of the containment, the reactor cavity, and the steam generators. Such decontamination can significantly reduce the amount of radwaste generated, help minimize the amount of protective clothing to be processed and reduced the area where respiration protection is required. Also, a reduction in worker exposure, time and cost can be achieved.

A safety review of the procedure was conducted which determined that the decontamination did not involve an unreviewed safety question. The determination is based on the following points and special conditions:

Since identification and protection of safety-related and/or EQ items is accomplished prior to decontamination, the containment decontamination does not impact the operability of those items and only pure demineralized reactor grade water is used in the effort so interaction with a chemical or other foreign material cannot result.

The introduction of unborated water under the conditions specified does not affect any safety related system, nor does it create the potential for unacceptable dilution of the primary system boron concentration.

Since the shutdown boron concentration in the reactor is maintained with margin at all times and the reactor head is in place during the steam generator decontamination, the procedure does not adversley affect the basis of any technical specification.

<u>Special conditions</u>

Generic to reactor cavity decontamination with the reactor vessel head less than fully tensioned and steam generator decontamination:

- assure RHR flow at least 400 gpm nominal.
- assure a 62.0 ft. water level is maintained.
- insure only demineralized reactor grade water is used.
- verify the RCS boron concentration is 3000 ppm at the start of decontamination.

Reactor cavity decontamination with the reactor vessel head less than fully tensioned.

- insure the reactor head is set in place with tygon tubing tetween the reactor head and vessel flange prior to decontamination.

- insure a mazimum total decontamination flow of 40 gpm from all sources within the reactor cavity.

- assure primary system is sampled and analyzed for boron concentration and appropriate adjustment made back up to 3000 ppm if necessary at least once every four hours during reactor cavity decontamination.

Steam generator decontamination.

- the decontamination of two channel heads (e.g. one hot leg and one cold leg) involving only one pass (maximum of 20 gpm per channel head for a maximum of 3 hours) is allowed at any one time. Prior to each additional decontamination the boron concentration must be readjusted to the initial 3000 gpm.

- 2000 gallons must be drained from each intermediate leg following completion of the associated decontamination and prior to fillup of the primary system. During this draining and subsequent system fillup 300 ppm boron concentration and water level will be maintained in the primary system.

- Sufficient communication shall be maintained to assure that all decontamination activities are stopped immediately should RHR flow be lost.

21 <u>Operation of CNSI module demineralization system for reactor cavity water cleanup, per EH&S procedure 4200 Rev. 1, and TPC 86-011</u>.

A brief description of the system and the contents of EH&S procedure 4.200 Re. 1 were given in Item 19 of this report. The specific application of the procedure for the reactor cavity water cleanup introduced certain differences in description and control measures specified i the temporary procedure change TPC-011.

The system was temporarily located in the Unit 2 containment. The suction from the demineralizer was located no more than 12 inches below the surface of the cavity water level. The discharge line was located no more than 6 feet below the surface of the cavity water level. Both lines were tied to a fixed structure on containment el. 95 ft. This ensured that the cavity water level was maintained at 23 ft. above the reactor vessel flange. The cavity level was checked once per hour and documented in a written log.

Spent resin in the vessels was sluiced to a High Integrity Container (HIC) in an OSSC storage container in the Unit 2 containment doghouse metal building. The roll-up door and the exit to the MOB was dammed off with a 6-inch high dam. The dammed area was large enough to hold the entire contents of the HIC in the unlikely event of a spill. The OSSC was also large enough to hold the contents of the HIC in the event of HIC failure.

Thus the use of this system was deemed not to involve an unreviewed safety question.

22. Addition of spare disconnect switch on the 6.9 Kv supply bus from the 13.8 KV/6.9Kv transformer.

A spare disconnect switch was added on the 6.9 Kv supply from the 13.8 Kv/6.9 Kv transformer for miscellaneous use.

The 6.9 Kv bus is not a safety-related bus and does not normally energize buses #5 and #6. Therefore the addition of the switch does not affect any safety related equipment.

23. Temporary repair of the sump grating on the recirculation sump.

The purpose of this modification was to temporarily repair the recirculation sump grating by adding steel caging mesh over the openings in the grating as described in the plant maintenance temporary repair procedure MP-4.16. Stainless steel binding wire was used to secure the mesh to the grating. This provided equivalent protection for accident performance pending completion of permanent repair.

24. Whip restraints for steam generator blowdown lines #45 through #48 past the isolation valves.

Whip restraints were installed on the steam generator lines #45 through #48 These are made of structural steel and their function is to protect the blowdown piping, containment penetrations and isolation valves against pipe whipping caused by a postulated failed high energy blowdown line.

The blowdown lines ran through the pipe penetration area which contains safety related equipment. To prevent any interaction between the blowdown line break, lines #45 through #48 were restrained against pipe whip.

The structural steel installed does not affect the seismic structures that is attached to. The installation enhances the ability to restrain a blowdown line rupture and cannot degrade the safe operation of any safety-related system.

25. Installation of a temporary manual "switch box" for waste gas analyzer sampling.

The waste gas analyzer (WGA) recorders and programmer were replaced. These components control the automatic sequencing of valves for samples to the waste gas analyzer. Prior to the removal of these components for replacement, a temporary manual "switch box" was installed to provide the sampling capability. The "switch box " was located approximately twenty feet from the WGA cabinet and was connected via a temporary cable to the rear of the WGA cabinet. The box contained 19 individual switches, which operate each valve independently. This would allow the operator to manually duplicate the removed component's function of automatically aligning these valves.

Four of the 19 valves associated with the WGA were containment isolation valves. The temporary modification did not affect the ability of these valves to close on a containment isolation (CI) signal since the upstream contacts would open deenergizing the circuits for the valves. If any of these valves would open via the "switch box" they would still close when the "switch box" would not override the condition that two independent operator actions are needed to open a CI valve after a CI signal. These two actions would be CI signal reset and the idividual valve reversion from manual control to automatic. Once these would be complete, a sample could be taken using the "switch box."

The temporary "switch box" is intended to replace the automatic WGA sequencing function for the period of time necessary to replace and test the WGA programmer and recorders. No accident analyses are affected by this change and the safe operation of systems components are not degraded.

26. <u>Replacement of RV-835 valve and/or spring to a 52 psig setpoint on the component cooling water surge tank</u>.

The setpoint of the component cooling water surge tank relief valve RV835 was reduced from 125 psig to 52 psig. This reduced the overpressure bounding value from 258 psig to 185 psig which is within the code allowable for the component cooling water system. This setpoint was changed to accomodate a system pressure transient due to a postulated rupture of an RCP thermal barrier cooling coil and concurrent failure of high return flow cutoff valve FCV-625. Thus, the system can still function as described in the FSAR.

The lowering of the setpoint still allows the component cooling system to be a closed system under accident conditions, even at 110% of containment design pressure.

27. <u>Power plant maintenance information system (PPMIS) computer cable network</u>.

This modification involves the installation of CRTs in various locations within the administration facilities and the central control room at Indian Point Unit 2. The conduit and cabling for the CRTs is also part of the modification.

The CRT located in the central control room is seismically restrained to ensure that it cannot affect any vital equipment in the area during an SSE. The existing penetration through the central control room wall which contains a fire barrier seal was used for the cable.

This system does not provide any safety-related functions and is installed such that no safety-related systems are adversely affected.

28. Pressurizer PORV control modification.

This change involved the disconnection of the PORV trip bistable PC-455F from the output of the programmed pressure controller PC-445K and the connection of it in parallel. This permits the bistable to directly sense the true pressurizer pressure. The 100 ohm input resistor for the PC-455F was removed since the bistable now senses a voltage signal. The modification assures that the PORVs function properly and open only when a true pressure reaches or exceeds the setpoint. Thus, the functioning of the PORV is improved, no other systems are adversely affected and no accident analyses are affected by this change.