Indian Point 3
Nuclear Power Plant
P.O. Box 215
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New York Power Authority

June 8, 1984
IP-LAH-183

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

Dr. Thomas E. Murley, Regional Administrator
Region 1
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

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

Dear Dr. Murley:

The following constitutes the annual report on changes, tests and experiments for Indian Point 3 Nuclear Power Plant as required by 10CFR50.59.

The Code of Federal Regulations, 10CFR50.59 (a) specifies that changes to the facility as described in the safety analysis report, changes in the procedures as described in the safety analysis report and conduct of tests or experiments not described in the safety analysis report may be made without prior Commission approval provided the proposed change, test or experiment does not involve a change in the technical specifications incorporated in the license or constitute an unreviewed safety question.

All the electrical modifications have been designed considering original separation criteria thus maintaining the integrity of electrical separation where required. These modifications were installed in accordance with standards equal to or better than those used during original installation. These modifications have been therefore deemed to not involve an unreviewed safety question.

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Any welding on the involved modifications has been accomplished using appropriate plant specific procedures based on applicable codes. These modifications were designed considering both thermal growth and seismic criteria as appropriate. They were also fabricated and installed in accordance with standards equal to or better than those used during orginal installation. These modifications have been therefore deemed to not involve an unreviewed safety question.

A description of such changes, procedures and tests performed at Indian Point 3 and a summary of the safety evaluations of each for the period of January 1, 1983 to December 31, 1983 are contained in Attachment I. Each has been reviewed 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 for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created, or the margin of safety as defined in the basis for any technical specification has not been reduced. It was concluded that the changes, tests and experiments do not constitute an unreviewed safety question.

Very truly yours,

John C. Brons

John C. Brons
Resident Manager

LAH:jd/02
Attachment

cc: Robert De Young, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Attn: Documents Control Desk

IP3 Resident Inspector's Office

ATTACHMENT I

Modifications and Evaluations

IE24
1/1

78-03-049 FW - Low Flow Feedwater Bypass Modification

The subject modification was installed to improve the low flow control of the feedwater system. In the original plant design, feedwater flow was regulated over its full range by four control valves arranged so that each valve controlled the flow to one steam generator. To promote finer control of low load feedwater flow, a bypass system was installed around each of the existing feedwater flow regulating valves. Small pressure sensing lines branching off the main steamlines were also added to provide a signal to new steam flow transmitters for monitoring low steam flow conditions.

Considering the small size of the pressure sensing lines, the probability of failure of the main steam system was not increased.

The new electrical installation for the control systems maintained the original logic of the system - i.e., there was no influence of the control system on the protection system.

All the tubing and parts associated with the modification which are subjected to main steam line pressure are category I, missile protected, have adequate separation for protection against pipe whip damage, and were designed with applicable seismic criteria.

80-03-054 - RMS Wide Range Gas Monitor for Plant Vent

This modification installed a Wide Range Gas Monitor system which monitors the plant vent effluent. This was performed in compliance with the requirements of NUREG-0578, Section 2.1.8b for increased range of radiation monitors. The system is capable of monitoring twelve (12) decades of effluent noble gas concentrations. It can be manually and automatically controlled. There are three ranges of indication with a minimum of one decade overlap between ranges.

Indication is provided by recorders installed in the control room supervisory panels. The Read-Out/Control Module located in the control room, gives additional control/indication of system parameters.

The system was channelized and power was supplied from a vital instrument bus fed from an inverter with a battery back-up. The entire monitor, including the Sample Conditioner, Wide Range Gas Detector, RM-80 Electronic Assembly and the Read-Out Assembly and Recorders in the control room, is designed to meet Class IE safety requirements and has been qualified to IEEE 323-1974 and IEEE 344-1975 standards. In addition, it meets the following qualifications: ANSI N13.1 - 1969 N320 - 1978, USNRC Reg. Guide 1.89, USNRC Reg. Guide 1.97 (Rev. 2).

80-03-062 ESS - "Bright is Right" Valve Indicating Lights

The purpose of this modification was to replace the existing "Bright is Right" valve position indication lights in the control room with an upgraded system. This system provides independent indication of safeguards and other important valve positions during plant normal and accident conditions. The old system used indicating lights which varied in brightness to show the valve position. Valve position was difficult to determine with that design. The new indicating system consists of two lights, one (red) for valve positioning and the other (white) for supervisory. The new indication lights clarify valve position and alleviate cause for confusion. Therefore the new lights are an improved human factors design which provide better indication to the operators.

The new indicators were placed in the same general area as the old indicators with minor modification performed to supervisory panels. This modification was designed in accordance with applicable seismic criteria.

82-03-027 RCS - Removal of RTD Loop Isolation Valves

The purpose of this modification was to eliminate the maintenance problems associated with the 2" loop isolation valves and all the drain and vent valves in the RTD bypass loops in the reactor coolant system.

The existing, isolation, drain and vent valves (numbers 561 A, B, C & D, 562 A, B, C & D, 563 A, B, C & D, 564 A, B, C & D, 565 A, B, C & D, 566 A, B, C & D, 567 A, B, C & D, 568 A, B, C & D, 569 A, B, C & D, 570 A, B, C & D) were removed and appropriate spool pieces were installed. The variable spring hangers associated with each RTD manifold were also removed.

When Indian Point 3 was designed, the wet-well type RTD's were installed in bypass loops paralleling the main coolant loops instead of being installed directly into the main coolant loops. These bypass loops had isolation valves so that when an RTD had to be replaced only the required bypass loop needed to be isolated and drained. If the RTD's were in the main loop, the entire RCS would have to be drained.

However, operating experience has shown that the maintenance problems associated with these isolation valves have provided more operational problems than they solved. Elimination of these valves removed the maintenance problem permanently. When an RTD is required to be replaced, the RCS will need to be drained down. Since there are spare RTD's in place, this drain down can normally be performed at a convenient time when other maintenance is also required. Therefore, this modification will upgrade the overall performance of the RCS. This modification was designed in accordance with applicable seismic criteria.

82-02-033 FW - Installation of Leading Edge Flowmeters (LEFM) in the Steam Generator Feedwater Lines

Leading Edge Flow Meters (LEFM) were installed in the steam generator feedwarer lines to provide more accurate measurement of feedwater flow which is used in the heat balance calculation to determine actual reactor power.

The LEFM is more accurate in its flow measurement compared to the old flow measuring arrangement. The original Venturi nozzle flow meters (FI-418, FI-428, FI-438 and FI-448) measured a pressure drop across a known restriction. This was converted to feedwater flow as part of the calorimetric power calculation, which defined reactor thermal power as the product of feedwater flow and steam generator enthalpy change. During plant operation, the Venturi nozzles became fouled with deposits which narrowed the restriction, leading to an inaccurately high measured feedwater flow and a correspondingly high thermal power calculation. This resulted in recalibration of the power detectors which led to the unit operating at less than its licensed power level.

The LEFM relies upon ultrasonic signal transmission for flow calculation. Transducers are set in the pipe pieces such that four upstream and four downstream pulses may be sent and received. The measured time difference between transmission and receipt, corrected for feedwater temperature and the speed of sound within the pipe, forms the basis for the flow calculation, which is done by a dedicated computer.

Pipe sections were removed between the feedwater regulating valves and the feedwater flow venturi nozzles. Then pipe sections containing transducers of the new LEFM were installed. The transducers were connected to a console which processes the transducers' signals and were linked with the plant computer for visual indication of feedwater flow for the reactor operators' use. The piping sections containing the transducers were of equivalent rating as the existing piping installed in the feedwater lines. Thus, the piping was not degraded. The piping has undergone a seismic and thermal analysis and has been found acceptable.

The modification allows the plant, via more accurate feedwater flow measurement, to operate at its licensed power level.

82-03-045 EL - TSC Cameras in CCR

The purpose of this modification was to install two cameras on the south wall of the control room in order to permit the control room panels to be observed in the Technical Support Center (TSC). The cameras were mounted on tilt/pan bases and possess zoom lenses which enable observation in the TSC of devices and indications on the supervisory and flight panels.

The two cameras supply the signal to a maximum of 4 monitors in the TSC. Control of the cameras for pan and tilt originate from the control console.

The mounting of the cameras, terminal boxes and conduit to the south wall of the control room was considered Category I. This modification was designed in accordance with applicable seismic criteria. The cameras are contained in a seismically qualified metal cage.

The penetrations through the control room floor and cable spreading room wall, which contain the rigid conduit from the control room through the cable spreading room, were sealed in order to meet existing plant fire protection design criteria.

82-03-066 EL - Modification to Control Room Emergency Lighting

This modification reduces the time required to accomplish the A.C. to D.C. transfer for the control room and other control building emergency lighting and increase the reliability of the A.C. supply to the control room emergency lighting.

In addition, this modifies the wiring of the overcurrent lockout protection relay associated with the D.C. control circuitry and the feed and tie breakers for lighting buses 32 and 33.

The transfer initiation relay is Time Delayed Normally Energized (TDNE). It is enclosed within transfer switch No. 33 which is an Agastat Time Relay which was set at seven seconds. This relay is de-energized by loss of normal voltage. After "timing-out" it allows relay "ER" to pick-up and energize the motor driven operating mechanism requiring approximately three additional seconds to complete the transfer. The setting for "TDNE" was reset to approximately two seconds, thus allowing the transfer to be completed in approximately five seconds.

The AC feed for control room lighting was originally supplied by lighting bus No. 32. The loss of power that initiates the transfer to emergency lighting was sensed on lighting bus No. 33. The AC feed for control room lighting was relocated from lighting bus No. 32 to lighting bus No. 33. Now the loss of power is sensed from the same bus that supplies normal control room lighting.

The overcurrent lock-out protection relay was a Westinghouse type MG-6. When activated, this relay tended to fail due to continuous energization conditions. To rectify this situation, the relay was modified such that it is automatically isolated from its voltage supply after it performs its function. To achieve this, a normally closed contact of this relay was electrically tied in series with the neutral terminal of the relay coil. The configuration allows the relay to pick-up and to open the voltage supply to the coil. When the relay is manually reset, as is the practice following an overcurrent condition, the circuitry to the coil of the relay is restored since the contact will close.

Bus (lighting transformer) loading was increased an additional 2400W (approx.). The addition of this load does not create an overload condition on lighting transformer No. 33. The load was attached in such a way as not to create a phase loading imbalance in excess of what presently existed before.

The installation of the two permanent test lamps preclude the previous practice of manually inserting test lamps in the Reactor Protection Racks during periodic testing of the Reactor Protection Logic Channels. These two test lamps (one for each logic channel) are located in the "Events Recorder" circuits which are retired and removed. The test lamps are "OFF" when the corresponding logic channel circuit is tripped. The event-recorder circuit incorporates this new component and has its own AC power supply and interfaces with the logic circuits at the breaker auxiliary contacts.

The installation of a second set of contacts in parallel with the existing contacts for the undervoltage (UV) trip relays fortifies the existing electric circuitry and prevents an inadvertent trip due to a single malfunction in a contact. Any malfunction in the dual contacts will not affect the operation of the "UV" relay. Any open circuit or grounding or short circuit will not compromise the operation of the "UV" relays. This installation enhances the reliability of the "UV" trip circuits and thus increases the margins of safety.

The installation of key operated switches is in the logic test rack in the control room. The placement of this switch in the "BYPASS" position will illuminate a light on the logic test rack, annunciate an alarm in the control room and place the reactor trip breaker being tested in the open position but physically racked in. Electrically this breaker is isolated from its closing and trip circuitry (the breaker will not physically change positions and the undervoltage trip mechanism will remain de-energized or in the trip position).

Also the bypass breakers 52/BYB (52/BYA) which were previously kept in the racked-out and open position are now kept in the racked-in position and locked open. A lock was installed on each bypass breaker which initiates a mechanical trip against each breaker not allowing them to close.

This installation reduces the wear on the reactor trip breakers and increases the breaker's reliability. It will decrease the probability of an inadvertent trip during the channel logic testing.

This modification does not affect that portion of the evaluation performed as required by 10CFR50 Appendix R.

83-03-007 COMP - Radial Flux Tilt Computer Alarm

The radial flux tilt alarms on the SB1 panel in the control room operate through the plant computer rather than the "NIS Detector Comparator". This results in a more reliable indication of flux tilt and prevents inadvertent actuation due to normal signal fluctuation.

As the computer uses time-averaged signals for its calculations, the alarm is shielded from signal fluctuation. Additionally, the computer is able to recognize when a power range detector is malfunctioning or out of service, and informs the operators accordingly.

A program has been written to perform the following functions: 1) input the power range values 2) determine whether all inputs are reliable, 3) calculate the flux tilt for each upper and lower quadrant, 4) compare the results to alarm setpoint and 5) actuate the alarm if necessary. At power levels below 50 percent, the alarms are set to annunciate, indicating "Auto Defeat". This is consistent with the existing alarm logic.

The cable from the existing "NIS Detector Comparator" to the alarm annunciators was disconnected. New cable was installed from the P-250 computer and reconnected to the same alarm points.

The tilt alarm system referred to in the Technical Specification is an upper/lower tilt alarm driven by the plant computer using the above technique. The previously existing alarm on panel SF ("Computer Alarms: NIS Rad Tilt or Rod Deviation"), also driven by the plant computer, serves as a backup. Both flux tilt monitors individually compare the upper detectors to each other and, separately, the lower detectors to each other.

The cable and conduit included as part of this modification was installed in accordance with channelization and separation requirements. The cable and conduit installation was classified as Cat. I in order not to compromise existing plant protection.

The modification incorporates a more stable determination of core quadrant power. This results in a more reliable calculation of radial flux tilt, with a corresponding decrease in spurious alarm actuations.

83-03-045 MTG - Main Turbine Spindle Replacement (LP-1 and LP-3) and Reblading (LP-2)

The LP Turbine Spindles No. 1 and 3 have been entirely replaced with a new upgraded disc keyway design and improved AISI 630 blade material. The keyways are designed to eliminate the problems with stress corrosion cracking encountered with the former design. The new blade material, AISI 630, is a much tougher material than the AISI 430 used in the blades prior to replacement. Also the geometry of the blades was altered to improve performance and reduce risk of failure at load.

Low Pressure Turbine Spindle No.2 was not replaced, however the L-1 and L-2 governor end and L-1 generator end blades were replaced. The new material (AISI 630) and geometry used in the manufacture of these blades is identical to the blades in Low Pressure Turbine Spindles No. 1 and 3.

The January 1983 Energy Systems Group Report entitled "Analysis of Hazards to Indian Point 3 Nuclear Generating Station Due to Missiles Generated by Turbine Failure" states, "The probability that turbine missiles would cause an offsite release of radioactive material calculated in this study is within the acceptance criteria of 10^{-7} per year as stated in Regulatory Guide 1.115, Rev. 1 "Protection Against Low-Trajectory Turbine Missiles".

As it can be seen from the evaluation presented, the new spindle keyway design and new blade material is an upgrade and will not decrease turbine reliability or operational safety.

83-03-077 MTG - Main Generator Radio Frequency (RF) Monitor

The Radio Frequency (RF) monitor is used to measure the RF emission from the arcs that may develop due to stator insulation failure. Early detection of a developing abnormality within the generator was the purpose of this modification.

In conjunction with the measurement and recording of RF level, it was necessary to record the megawatt output of the main generator. The megawatt and RF levels are correlated to each other as well as the time rate of change of these levels.

The RF monitor is comprised of a radio noise meter and a recorder that contains the necessary electronics to generate a High/Low alarm for the annunciation of RF problems. The equipment is located in a cabinet in the Turbine Building with the alarm located in the control room.

A cable was installed from the monitor to the RF voltage probe. The voltage probe was installed and power for the monitor was supplied from Lighting Panel No. 32. Cable will be installed from the Conitel Cabinet to the RF monitor in order to provide the megawatt indication of the main generator at a later date.

The portion of this modification that involves cable routing in trays was defined as Category I and was, therefore, installed in accordance with the appropriate procedures for maintaining the site electrical separation criteria. All other portions of the modification were non-category I. The installation of the RF monitor was consistent with existing generator fault detection and therefore did not increase the probability of a failure causing a unit trip or prevent a present protective device from performing its function.

83-03-088 SG - Metal Impact Monitoring for the Steam Generators

This modification relocated the metal impact monitoring detectors for each steam generator to permit monitoring both the primary and secondary sides of the tube sheet. Previously only the primary side was able to be monitored.

Metal impact sensors were installed in holes drilled and tapped in each steam generator primary and secondary side shells. Each hole is approximately one-half inch deep and a quarter inch in diameter. Terminal boxes for each sensor were mounted to the existing steam generator support structure.

The Category I portions of this modification were the eight holes drilled into the steam generators. An analysis performed by Westinghouse has determined that the addition of the sensor holes did not reduce the integrity of the steam generator shells for all steady state, transient and accident conditions. The eight terminal boxes were mounted on unistrut and fastened by metal clamps to the existing steam generator support structure. This modification was designed in accordance with applicable seismic criteria.