10 CFR 50.59(b) REPORT OF CHANGES, TESTS AND EXPERIMENTS 1987 THROUGH CYCLE 8/9 REFUELING OUTAGE

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT UNIT 2

Docket No. 50-247

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Preface

Discussed herein are descriptions and safety evaluation summaries of changes, tests and experiments performed at Indian Point Unit 2 and completed in 1987 and through the Cycle 8/9 Refueling outage. These have been evaluated and determined to meet the following criteria as established by 10 CFR 50.59:

Criteria

1. 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.

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.

It has, therefore, been concluded that none of these changes represents an unreviewed safety question.

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1. Cycle 8/9 Refueling - Cycle 9 Operation

Indian Point Unit 2 was refueled following the end of its eighth cycle of operation. A total of 68 fuel assemblies were removed and replaced with low-parasitic (LOPAR) type assemblies enriched to 3.4 and 3.7 percent. The core also contains 56 new wet annular burnable absorber (WABA) assemblies with clusters of 4 and 16 rods, giving a total of 800 fresh WABA rods. The new core has a low neutron leakage loading pattern similar to that of the three previous cycles.

The mechanical design of the fresh region of fuel loaded at the start of cycle 9 is the same as earlier regions except that the fuel pellets have a small chamfer at the ends, the pellet holddown springs exert less force, the grid sleeve uses a low-carbon material, and the bottom end plug has a radius (bullet nose). For cycle 9 only, unpressurized WABA assemblies were installed. These differences do not result in any significant change in core performance or safety characteristics.

A report entitled "Reload Safety Evaluation, Indian Point Nuclear Plant, Unit 2, Cycle 9" was prepared by Westinghouse Electric Corporation. The report presented an evaluation for Cycle 9 which demonstrated that the core reload, including the design changes discussed above, 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 new reactor core design, and it was concluded that acceptable safety limits will not be exceeded for any incident.

These changes do not involve an unreviewed safety question and the Technical Specifications are not affected.

2. Reactor Vessel Closure Stud Replacement

Stud tensioners are used during installation and removal of the nuts which retain the reactor vessel closure head. The tensioners had been threaded onto the studs, but the tensioners and studs have been changed to use a quick disconnect coupling device. The reactor vessel closure studs were replaced with new studs having grooves to mate with the tensioners. The new and old studs are identical in material, size, weight, fits, and tolerances. All original design criteria are met except that the replacement studs are not N-stamped, which is allowed by Article IWA 7210(c) of the ASME Boiler and Pressure Vessel Code. The requirements of the Code are satisfied. No unreviewed safety questions are involved with this change.

3. Instrumentation Port Column Modification

The instrumentation port column assembly is a pressure-retaining seal through which the thermocouple leads penetrate the reactor pressure boundary. The original assembly had a large number of parts and a time consuming installation procedure, which resulted in significant radiation exposure. The assembly was replaced with a simpler assembly utilizing articulated clamps to seat the two conoseal joints. The installation and disassembly is much easier and will result in decreased personnel radiation dosage. The assembly has been designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, division I, including normal operation, seismic and accident conditions. This change does not involve a change in the technical specifications or their bases, and there is no unreviewed safety question involved.

4. Installation of Reactor Head Shielding

A new head shielding system has been provided to replace the previous one. The head shielding is used only during outages to increase the speed and efficiency of work in the reactor vessel head area and reduce radiation exposures to the personnel doing the work. The modification was determined not to affect the seismic integrity of the reactor head lifting rig and does not introduce any failure modes that could impact the safety related equipment in the area. This modification does not affect the bases for any technical specification and does not involve an unreviewed safety question.

5. Replace Hoist on Spent Fuel Handling Bridge

The hoist on the spent fuel handling bridge is used to lift and move fuel assemblies in the spent fuel pool. It has been replaced with a new hoist which increases safety of fuel handling operations. The new hoist has a double reeving system that allows one wire rope to hold the load if the other rope fails. Other changes included improved load monitoring capability, provision of two lifting speeds, means for emergency lifting and lowering by hand, and inclusion of both electrical and mechanical load brakes. These changes all enhance safety, and do not involve an unreviewed safety question nor have an adverse bearing on the technical specifications.

6. Control Rod Drive Mechanism Ventilation Fan Replacement

The four control rod drive mechanism ventilation fans and motors were replaced, due to the high failure rate of the motors. The replacement motors have a higher horsepower rating. The aluminum fan blades have an accepted protective coating. The fans and motors are seismically mounted from the missile shield, and use the existing power supplies and cables. The horsepower increase does not affect the motor control center load requirements, nor does it affect diesel generator loading since this MCC is stripped on automatic start of the diesel generators. These modifications do not involve the bases of a technical specification or an unreviewed safety question.

7. Reactor Head Cooling Shroud Inspection Opening

To facilitate visual inspections inside the cooling shroud, one inspection opening was made. The opening is equipped with a hinged cover that is bolted shut during operation of the reactor. The new opening does not change the structural or seismic integrity or function of the shroud. The cover is of the same material as the shroud. No change to the technical specifications and no unreviewed safety question are involved with this change.

8. Provide Shielding for Lower Reactor Internals During In-service Inspection

In order to reduce radiation exposure of personnel performing the reactor pressure vessel in-service inspection during refueling outages, a shielding system has been provided for the lower reactor internals. It is only used after the reactor has been defueled and the lower internals package placed in its laydown area. The shielding system is not lifted over the reactor while the reactor contains fuel. The structural integrity of the reactor cavity reinforced concrete is not affected by installation of the shielding. This change does not involve an unreviewed safety question.

9. Removal of Pipe Whip Restraint PWR-124

Pipe whip restraint PWR-124 on the pressurizer surge line has been removed to provide additional working space on the platform between steam generators 23 and 24. The surge line piping was analyzed and the restraint was removed in accordance with NRC Branch Technical Position MEB 3-1 of Standard Review Plan section 3.6.2, and requirements of Generic Letter 87-11. The technical specifications and bases are unaffected by this change and no unreviewed safety question is involved.

10. Replace and Relocate Auxiliary Pressurizer Spray Valves 211 and 212

Valves 211 and 212 in the auxiliary pressurizer spray line of the Chemical and Volume Control System have been replaced and, along with part of the piping, relocated. Drain valve 55 has been removed. The valves were relocated to a lower radiation area to reduce the radiation exposure associated with surveillance, operation and maintenance activities. The new design is in accordance with appropriate criteria, is functionally equivalent to the previous design and has been re-analyzed to assure compliance with seismic requirements. Therefore, this change does not involve an unreviewed safety question. No change in the technical specification is involved.

11. Temporarily Replace Power Operated Relief Valve Nitrogen Supply Relief Valve with Two Relief Valves

Each pressurizer power operated relief valve (PORV) has an accumulator which stores nitrogen for its operation. A single relief valve protected the PORV operator from overpressure in case of failure of the pressure regulator between the accumulator and the PORV operator. The single relief valve had a flow area of 0.44 square inches and was replaced by two valves with a total flow area of 0.4 square inches. The original relief valve was plugged in place.

The new relief valves have adequate relief capacity to protect the PORV operator from overpressure. Their design precludes the potential for missile generation by plug ejection, and operability is not affected by seismic events. Thus, there is no unreviewed safety question. The technical specifications and bases are not affected by this change.

12. Generic Replacement of Material for Reactor Coolant Pumps No. 1 Seal

Aluminum oxide had been used in the manufacture of seal #1 for the reactor coolant pumps. Silicon nitride is now used, as it resists rubbing degradation at startup, and is otherwise equal or superior to aluminum oxide. The silicon nitride material has been shown to be suitable for the application based on extensive engineering, field testing, and in-plant service. No unreviewed safety question is involved in this change.

13. Reactor Coolant Pump Motor Oil Gooseneck Vents

To assure accurate oil level indications for the reactor coolant pump (RCP) motors under all conditions, the level devices have been locally vented to the atmosphere. They were previously vented through a long and convoluted path which gave erroneous readings under certain conditions. The vent is through a gooseneck located above the oil level, to prevent spillage. Should any oil be spilled, it will be caught and drained away by the RCP oil collection system. This change created no unreviewed safety questions.

14. Modify Wide Range Resistance Temperature Detector Connections

The splice connections to the wide range resistance temperature detectors have been environmentally qualified by providing an RTV-filled vapor-tight condulet enclosure. Seismic calculations show the additional weight is acceptable. This change does not involve an unreviewed safety question.

15. Reactor Coolant System (Narrow Range) Resistance Temperature Detector Replacement

Four new Weed resistance temperature detectors (RTD) were installed in the Reactor Coolant System in locations previously occupied by Rosemont RTDs.

Their performance is being monitored to verify their suitability for use as wholesale replacements for the existing detectors in the reactor protection system. They will not be connected to the reactor protection system until evaluation is completed. The RTDs have been tested to ensure they meet the performance requirements of Class 1E functions under design basis event conditions. Their design is consistent with Seismic Category I design requirements and their pressure rating, design and testing are consistent with reactor coolant system boundary requirements. This modification does not involve an unreviewed safety question.

16. Steam Generator Manway Insert Passivation Test

To test for the effect of surface condition on radioactivity buildup on out-of-core components, the manway insert (diaphragm) of steam generator #23 has been given three different surface conditions. They are as-built, electropolished, and electropolished and passivated. This test does not cause any functional change, does not affect the pressure boundary and does not introduce new materials into the coolant system. Therefore, it does not involve an unreviewed safety question.

17. Installation of Safety Injection Time Delay Upon High Steam Flow/Low T Average or Low Steam Generator Pressure

One of the circuits that initiates safety injection is high steam flow in coincidence with low T average or low steam pressure. The high steam flow setpoint is automatically adjusted based on plant load using a signal derived from first stage turbine pressure. When the turbine trips, the first stage pressure and the load signal decrease very rapidly. The steam flow signal decreases less rapidly, which has caused spurious safety injection initiations. To reduce their number, a time delay of less then two seconds has been installed in the circuit for high steam flow coincident with low T average or low steam pressure. The only accident that results in safety injection from these conditions is the large steam line break outside of containment. The FSAR analysis included a four second delay for safety injection initiation from high steam flow and low T average, and a two second delay for initiation from high steam flow and low steam pressure.

Re-analysis shows that with an additional delay of two seconds, the acceptance criteria for this accident are met. Therefore, no unreviewed safety question is involved, and this change will reduce the potential for spurious challenges to safety systems.

18. Generic Replacement of Safety Injection System Pump Seal Water Coolers

The seal coolers and seal water piping-tubing assemblies for the safety injection pumps are being replaced on an "as-required" basis.

The replacement coolers are functionally equivalent but are somewhat more compact than the originals. The new coolers will be mounted differently but will still be capable of withstanding the safe shutdown earthquake and continuing to function.

The pumps and coolers are connected by an assembly which utilizes both piping and tubing components. This will be replaced by an all-tubing assembly which is functionally equivalent and meets the system requirements and specifications.

There is no change to the technical specifications, and no unreviewed safety question is involved in this change.

19. Accumulator Level Instrument Replacement

Accumulator level transmitters were removed and new ones installed at different locations. The new transmitters, which are from a different manufacturer, meet all requirements and qualifications of the previous transmitters. Electrically, this change only affected the wiring from the splice boxes to the transmitters, it did not affect any setpoints or alarms, and does not interfere with the capability to meet accumulator level technical specifications. These changes do not involve an unreviewed safety question.

20. Automation of Phase A Containment Isolation on Manual Safety Injection Actuation

Phase A containment isolation closes "nonessential" containment isolation valves, which are those which do not increase the potential for damage to in-containment equipment when isolated. The isolation circuitry has been changed to actuate a complete Phase A isolation (including ventilation isolation valves) on either automatic or manual initiation of safety injection. Prior to this change, the isolation circuitry closed only the ventilation isolation valves when safety injection was manually initiated, and actuated a complete Phase A isolation when safety injection was automatically initiated. This change was a result of the control room human factors review (DCRDR) and its system function and task analysis. This change reduced the chances for operator error, improved the reliability of the automatic containment isolation, and does not involve an unreviewed safety question.

21. Proteus Computer Connections

Connections were made to the Proteus (plant) computer to provide data from reactor coolant flow and reactor vessel level sensors. The coolant flow data is used in calculating reactor power for calibration of the Nuclear Flux Power Range Channels. Reactor power data is verified and validated prior to calibration of the power range channels. Reactor vessel level data is provided for evaluation of potential accidents. Connections were also provided to initiate control room alarms on computer room high temperature and on computer failure. The modification does not affect any plant safety equipment but only provides additional data, does not affect the plant technical specification requirements and no unreviewed safety question is involved.

22. Modify Fan Cooler Unit Alarm in Central Control Room

When a containment fan cooler unit is turned off, the normal operation discharge damper is closed to prevent backflow through the unit. This causes category alarm "Common Containment Recirculation Air Unit Valve Tripped" in the Central Control Room to be on when there is, in fact, no problem. To eliminate this nuisance alarm, the alarm circuit for each fan cooler unit was interlocked with the control switch. This is a "normal" flow alarm and is not involved with the accident low flow alarm. The alarm circuit is isolated from the control circuit. This change does not affect the technical specifications and does not involve an unreviewed safety question.

23. Generic Reconnection of Rod Position Indicators to Use Spare Cables

When defective cables or connectors are identified between the reactor vessel head and the rod position indicator cabinet, existing spare cables are substituted. The spare cables are identical to the normally-used cables, but may utilize a different penetration. There is no effect on system operation. This change does not involve an unreviewed safety question.

24. Monitor Turbine Trip on Reactor Trip Circuitry

This change provided indicator lights to allow the detection of any malfunction of the circuit which initiates a turbine trip on a reactor trip. The new installation does not change any of the functions of the system, and conforms with system design criteria and standards. There is no change to the technical specifications and no unreviewed safety question is involved.

25. 480V Breaker Control Circuit Modifications

The four diesel generator output breakers and three 480V tie breakers had lockout devices and trip bar switches which did not provide additional protection to the breakers and may misoperate under test conditions. The lockout devices and trip bar switches were removed from the output breakers.

Additionally, breaker misalignment can cause incomplete makeup of slide-type control circuit connectors in the output and tie breakers, and in four supply breakers. To improve breaker operation and system reliability, the slide connectors for all output and supply breakers were paralleled where feasible. The seismic capability of the breakers was not affected by this change and their functions were not modified. There is no FSAR impact. Therefore, the changes do not involve an unreviewed safety question.

26. Power Connections for SAS Signal Circuit Isolators

This modification extended the 118 VAC instrument bus supply to spare terminal points for future installation of Safety Assessment System Isolators. The configuration conforms to previous functional requirements, design criteria and standards. No unreviewed safety question is involved.

27. Battery Banks No. 21 and 22 Rack Ground Installation

To assure personnel safety, a cable has been added to ground the support racks for batteries 21 and 22 to the building steel. The cable is seismically restrained and will not affect the capability to withstand a safe shutdown earthquake. The change does not involve any technical specification or its bases, and no unreviewed safety question is involved.

28. **Replace Existing Single Conduit for Positive/Negative Side of Battery Nos. 22, 23 and 24 with Separate Cable/Conduit for Each Leg**

Both positive and negative feeds from each station battery were run together through a single conduit between the batteries and the DC panels, and the connection between the two sections of each battery bank were run in a single conduit. The single conduits were replaced with separate cable and insulated cable supports inside the battery rooms and separate conduits at panel entries. Transite barriers were also added to the fuse boxes between the positive and negative side fuses. The installation meets Seismic Class I requirements and the electrical portions are Class 1E. The technical specifications and bases are unaffected by this change and it does not involve an unreviewed safety question.

29. **Alternate Offsite Power 20 MVA 13.8/6.9 kV Auto Transformer Load Monitoring**

An ammeter has been physically mounted in the Central Control Room to allow the load current from the Gas Turbine substation (an alternate offsite power source) to be monitored in the future. The ammeter is seismically mounted, and seismic capability of plant systems is maintained. These changes do not involve an unreviewed safety question.

30. **Installation of 13.8 kV/480V Substation Outside the Indian Point Unit 2 95 Ft. El. Containment Hatch**

A non-safety related 13.8kV/480V substation has been installed outside the Indian Point Unit 2 containment hatch, with feeders from the retired Unit 1. The substation will only provide power for maintenance activities during outages, and will be connected to power boxes inside containment only when the permanent equipment hatch is removed. The circuits are electrically and physically isolated from safety-related circuits. This change does not involve an unreviewed safety question.

31. Motor-Generator Set Field Flash Circuit Modification

The motor-generator sets provide power for operation of the control rod drives. Their field flash circuits have been relocated from the Class 1E side of their power supply to the non-class side, thereby providing additional assurance that these non-safety circuits will not interact with the safety-related power supply. A double-pole switch was used to assure field flash circuit faults will be isolated during normal operation. No unreviewed safety question is involved.

32. Installation of Temporary Restraining Bar on 480V Bus Breaker

A restraining bar was installed on the breaker which feeds 480V Bus 5A from off-site power sources as a temporary substitute for the breaker's locking mechanism. The restraining bar assures that the breaker remains in place during normal, accident and safe-shutdown earthquake conditions. The bar is functionally equivalent to the breaker's locking mechanism, does not involve an unreviewed safety question and does not affect the technical specifications or their bases.

33. Temporary Removal of Transfer Bar from Lighting Transfer Switch

Transfer switch No. 24 allows power for some of the plant lighting to be obtained either from 480V bus 3A or from 480V bus 5A. The transfer bar prevents simultaneous closure of feeds from the two busses, which would interconnect the two redundant power supplies. Technical Specifications require that redundant power supplies not be interconnected.

Because the normal feed breaker would not close, the transfer bar was removed and the alternate feed breaker was closed. The normal feed breaker (which could not be closed) was tagged out in the open position to administratively prevent its closure, and a bracket was installed to provide a physical barrier to its closure.

Since alternate means were provided to perform the function of the transfer bar, the change did not involve an unreviewed safety question or a change in the technical specification.

34. Diesel Generator Differential Pressure Controller Test Tee Connections

The diesel generator units each have a duplex filter in the fuel system, with a differential pressure controller across each filter. To allow the differential pressure controllers to be maintained and calibrated without removing the diesel generators from service, shut-off valves and test tees have been installed. The test tees are normally closed (capped). Diesel Generators 22 and 23 have been modified.

This modification does not functionally change the system, does not degrade the Seismic Category I classification of the system and does not involve any technical specification or unreviewed safety question.

35. Place Valves Controlling Cooling Water to Emergency Diesel Generators in Full Open Position

The valves (FCV-1176 and FCV-1176A) which control the flow of service water for cooling the emergency diesel generators were temporarily placed in the fully open position and their flow control signals disconnected. The safeguards position for these valves is full open which provides maximum flow of cooling water. This jumper is intended to compensate for an unreliable flow control system until a new system is installed. A three-way temperature control valve prevents lube oil overcooling during shutdown to maintain the rapid loading capability of the diesels. This change does not involve the technical specification or its bases, and does not involve an unreviewed safety question.

36. Diesel Generator Cooling Service Water Discharge Valve Modification

Diesel Generator Service Water Control Valves FCV 1176 and 1176A were replaced with similar Jamesbury valves. The new valves were fabricated from Monel and are seismically qualified. The replacements do not involve any unreviewed safety questions.

37. Replace Emergency Diesel-Generators Service Water Inlet Valves

The six gate valves in the service water inlet lines to the Emergency Diesel-Generators were replaced with butterfly valves. The replacement valves are functionally equivalent to the previous ones. A seismic analysis was performed for the piping with the new valves to verify its adequacy. This change does not involve an unreviewed safety question.

38. Replacement of Service Water Piping to Instrument Air Compressors

Piping was installed between the 24 inch service water mains and the instrument air compressors, consisting of two 3-inch supply lines, one 3-inch return line and one 3-inch underground spare line. The original piping was retired in place. The new piping installation meets the design requirements for the service water system, including seismic criteria. The seismic qualification of the Control Building is not affected by the installation of penetrations for the new piping. This replacement is not involved with any technical specification or its bases, and does not involve an unreviewed safety question.

39. Temporarily Secure the Service Water System

In order to replace an unisolable section of the service water return line, the service water system was secured and the service water pumps and the emergency diesel generators were prevented from automatically starting. The reactor core was unloaded with the fuel in the Spent Fuel Pool during these operations. Loss of the heat sink for the Spent Fuel Pool and loss of offsite power are considered in the FSAR, and procedures existed to ensure that forced decay heat removal could be re-established within the parameters of the FSAR analysis, with an adequate margin. Therefore, no unreviewed safety questions were involved, and there were no changes to the technical specification required.

40. Procedure to Support Repair of a Component Cooling Line

A freeze seal was used to allow isolation and repair of an otherwise unisolable "dead ended" line in the Component Cooling Water (CCW) System. Examination of the affected portion of the piping and evaluation of the freeze process indicated no potential damage. An existing abnormal operating procedure which addresses loss of component cooling water covered the worst type of credible accident that could have occurred during the process. Therefore, no unreviewed safety question is involved in this change. All applicable technical specifications were met during the procedure.

41. Repair of Component Cooling Line 515

A damaged section of line 515 in the Component Cooling System was cut out and replaced. The replacement section (a length of 2 to 3 inches) is Schedule 80 pipe, while the original sections were Schedule 40. The new section exceeds the requirements for the service, and, due to its short length, the heavier wall does not affect the seismic analysis. In addition, since this line is now "dead-ended", it has no functional requirements. There is no unreviewed safety question involved in this change.

42. Change of Residual Heat Removal Pump Motor and Shaft

This change allows the use of a shaft made of stainless steel type 410 in a replacement motor of a modified design. The shaft material has been demonstrated to be satisfactory. The replacement motor is electrically almost identical to the original, requiring only a minor adjustment of the instantaneous protective relay setting. The maximum load is less than that assumed in the FSAR for the diesel loading analysis. The ability of the pump to meet its technical specification operability requirements is not affected. This modification does not involve an unreviewed safety question.

43. Monorail for Residual Heat Removal Pumps

A monorail and supporting steel have been installed outside the Residual Heat Removal Pump Rooms to facilitate lifting and moving the pumps for maintenance purposes. The installation is seismically qualified and the additional weight of the monorail and the loads it may carry have been analyzed and are within the structural capabilities of the supporting walls. The physical replacement of the RHR pumps and motors is not addressed in the technical specification, therefore, no change is involved. This replacement does not involve an unreviewed safety question.

44. Residual Heat Removal Pump Tests with Reactor Defueled and with Reactor Fueled

To better define operating conditions in the draindown condition, tests were run to confirm acceptable RHR pump and system parameters at various RHR flow rates and Reactor Coolant levels. The first series of tests was performed using one pump with the reactor defueled. Pump operability was then verified, the reactor core reinstalled, and a second series of tests performed. The test procedure and evaluation considered the potential for boron dilution and loss or degradation of reactor coolant flow in the RHR loop, and considered the effects of changing reactor coolant temperature. Each of these events can occur independent of this test, and appropriate recovery procedures are available. Therefore, the tests did not involve an unreviewed safety question or a change to the technical specifications. The test results will be used to revise RHR operating procedures.

45. Residual Heat Removal Loop Purification Pump

The Residual Heat Removal Loop Purification Pump has been replaced. This pump is used only to clean up the water in the reactor cavity during outages when the reactor is depressurized. It is designed to be compatible with the connected systems. This change does not involve an unreviewed safety question.

46. Fire Protection for Residual Heat Removal Pump No. 22 Power Cables in Residual Heat Removal Pump No. 21 Room

The feeder cables for Residual Heat Removal (RHR) Pump No. 22 are in a conduit routed through the cubicle for RHR Pump No. 21. To prevent loss of both RHR pumps from a fire in the cubicle for pump #21, this conduit was fire wrapped in the cubicle. The fire wrap does not affect either the seismic or the functional capability of the conduit. The technical specifications are not impacted and no unreviewed safety question is involved.

47. Replace Motor Operator for Emergency Boration Valve No. 333

The motor operator on valve 333 in the Chemical and Volume Control System was replaced. Concentrated boric acid is supplied from the Boric Acid Storage Tanks directly to the charging pumps through this valve. The new motor operator is identical to the old in fit and form, but the valve stroke time was changed from 10 seconds or less to approximately 35 seconds, which is within FSAR parameters. There is, therefore, no unreviewed safety question.

48. Replace Sockolet with Nipolet at Charging Pump No. 22

During repairs to the piping associated with charging pump #22, a sockolet fitting was replaced by a nipolet and a coupling. The replacement fittings are of the same material, size, and pressure rating as the sockolet, are functionally equivalent, and are designed to applicable codes. There is no change to the technical specifications, and no unreviewed safety question is involved.

49. Interim Operation without Primary Water Storage Tank Bladder

The Primary Water Storage Tank was equipped with a rubberized bladder which had been installed to separate the surface of the stored water from the atmosphere, thereby preventing oxygen intrusion into the tank contents. Upon inspection, the bladder was found to be torn beyond repair. It was therefore removed, with operation of the Primary Water Makeup System continuing until a new bladder could be obtained and installed. The only effect on the system is that, in the absence of the bladder, atmospheric oxygen dissolves into the stored water. This additional dissolved oxygen is controlled by the addition of hydrogen to maintain RCS oxygen limits within specification. Besides providing demineralized water for RCS makeup blending, the primary makeup system serves the safety related functions of makeup to the Component Cooling Water Surge Tank, emergency makeup to the Spent Fuel Pool, emergency cooling of the Safety Injection Pumps and the Residual Heat Removal Pumps, and makeup to the Isolation Valve Seal Water System Tank. None of these functions is impacted by the absence of the bladder since deaerated water is not required. Consequently, the change does not involve an unreviewed safety question. The system is not controlled by the technical specifications.

50. Install Temporary Pipe Clamp Assembly on Residual Heat Removal Miniflow Valve

A temporary pipe clamp assembly was installed on valve 1870 to improve valve stability during pump startup for tests. This change does not involve an unreviewed safety question since the clamp does not affect valve operation, system design functions, or seismic capability, nor will it become a missile during a safe shutdown earthquake. This change does not affect the technical specifications or their bases.

51. Install Temporary Clamps on Letdown Isolation Valves 201 and 202

A temporary clamp assembly was installed on valves 201 and 202 for a maximum period of six months. These valves are the containment isolation valves in the Chemical and Volume Control System letdown line. The valves were repaired and the clamps removed during the Cycle 8/9 refueling outage.

The clamp assemblies were provided as additional assurance of bonnet integrity since the bonnet bolting had indications of corrosion, and were designed to provide body-to-bonnet-holddown capability equivalent to that of the original valve design. Materials are compatible with those of the valve and the piping. The clamp will not interfere with valve operation, even under safe shutdown earthquake (SSE) conditions, will not affect the ability of the line to withstand SSE and maintain its functional capability, and will not become a missile in the event of SSE. Therefore the installation of the clamp did not involve an unreviewed safety question or a change in the technical specifications or their bases.

52. Repair of Girth Weld in Steam Generator No. 22

During an ultrasonic inspection of the circumferential girth weld between the transition cone and the upper cylindrical shell of steam generator #22, indications of cracking in the vicinity of the weld were observed inside the shell. Corrosion pits were also observed inside the shell.

Fabrication and inspection records of the weld did not reveal any materials, conditions or circumstances which could have caused or predicted the conditions found. The current understanding is that no single mechanism is responsible for both initiation and propagation of the indications. Corrosion pits are postulated to have developed during early operations when on occasion, the secondary water contained relatively high oxygen concentrations and other contaminants such as chloride and copper ions. The pits could have acted as nucleation sites for cracks which, once initiated, were then propagated by a combination of factors including corrosion, static stress, and corrosion fatigue.

The weld was repaired by grinding out the cracks. The final configuration of the grindouts and the acceptance standard for the significant subsurface indications were determined based on stress analysis and comparison to flaw size standards in the ASME Code. A stress analysis and fatigue usage evaluation using the rules of the ASME Boiler and Pressure Vessel Code was performed and the repair was found to be acceptable. Consistent with the FSAR, the as-repaired shell of the steam generator conforms to the ASME Code, Section III, Class A requirements. Although the grindouts resulted in local wall thicknesses less than the nominal wall thickness stated in the FSAR and the steam generator design drawings, such locally thinner areas are acceptable using the rules of the ASME Code.

Therefore, it was concluded that the above described repair of Steam Generator 22 did not result in an unreviewed safety question.

53. Install Sentinel Plugs in Two Steam Generator Tubes

Sentinel Plugs have been installed on both hot and cold leg sides of two steam generator tubes. The tubes did not require plugging because of leakage or eddy current test indications, but were plugged as a precaution against failure by a fluid-elastic vibration phenomenon because a positive anti-vibration bar holding indication was not available.

The sentinel plugs have a 13.5 mil hole through the center, which will limit the primary to secondary leak rate to approximately 0.35 gpm in the event a sentinel-plugged tube experiences a double ended rupture. Since the leak rate would exceed the Technical Specification limit for normal plant operation, shutdown and investigation/repair would be required. Interaction between the ruptured tube and active steam generator tubes would be limited, precluding multiple tube ruptures.

The sentinel plug meets the design requirements of the ASME Boiler and Pressure Vessel Code and other applicable requirements for steam generator tube plugs. The installation of the sentinel plugs does not involve a change in the technical specifications or an unreviewed safety question.

54. Steam Generator Primary Manway Closure Studs Hydraulic Tensioning System

The manways on the steam generators are retained by studs and nuts. To facilitate the use of a stud tensioning device for installation and removal of the manways, the studs were replaced with new studs that are 3 1/2 inches longer. The replacement studs meet the same material specifications as the previous studs. Therefore, no unreviewed safety question exists.

55. Replacement of Main Steam Isolation Valve Disc Stop for MS1-21 through 24

This change is to allow use of a low-carbon material when changing the disc stop of main steam isolation valves MS1-21 through -24. The low-carbon material does not require post-weld heat treatment, but has the same Code-allowable stresses as the old material. Existing calculations are, therefore, still valid and the change does not involve an unreviewed safety question.

56. Main Electrical Generator Replacement and Installation of Accessories and Supervisory Equipment

The main electrical generator has been replaced and supervisory equipment and accessories have been replaced and modified as required by the new generator.

This equipment is not safety-related. Impacts on safety-related equipment and systems have been reviewed to assure there are no effects on their functional capabilities, and that the new equipment and installations meet applicable requirements (seismic, fire protection, etc.). This change does not involve an unreviewed safety question.

57. High Pressure Turbine Rotor Replacement

The high pressure turbine rotor has been replaced with a new rotor designed to fit into the existing casing. The new rotor has higher mechanical strength than the one it replaced and is optimized with respect to electrical output for the current plant power level. Previous FSAR analysis has shown that the high pressure turbine was not a credible source of missiles, and this analysis also applies to the new rotor. The turbine does not perform any safety or mitigating function. No technical specification is impacted and no unreviewed safety question is involved.

58. Electrical Generator Torsional Tests

Tests were conducted on the turbine-generator unit as currently configured (following generator replacement) to identify its natural torsional tendencies. A single-phase torsional excitation test was performed which required disconnection of the generator from the electrical grid, use of a low-power excitation source instead of the normal source, grounding of one phase of the step-up transformer output, and varying the rotating speed between 100 rpm and 1965 rpm. Overspeed protection was provided by the Independent Electrical Overspeed Protection with its setpoint temporarily adjusted to 111% (1998 rpm) of normal operating speed, and by a dedicated operator who was to trip the unit if the speed reached 2020 rpm. A transient test was also performed in which the generator, in its normal operating configuration, was connected to the grid while out of phase by a small angle (5 to 10 degrees) within the normal tolerance. Evaluation of the test conditions showed that the potential maximum overspeed was less than the overspeed calculated for FSAR Appendix 14A.3. Therefore, the tests did not involve an unreviewed safety question.

59. Replace Obsolete Main Steam Flow Transmitters

The existing main steam flow transmitters are obsolete and have been replaced with new transmitters. These transmitters generate signals used to initiate protective actions in the event of either loss of feedwater or steamline break. Minor circuit changes were required to compensate for differences between the old and new transmitters, however, system function is unchanged except for response times. Analysis demonstrated the slightly longer response time (a fraction of a second) is acceptable. The replacement transmitters are environmentally qualified and meet all other requirements for this function. No unreviewed safety question is involved.

60. Disarm Low Pressure Steam Dump

The low pressure steam dump system consisting of six dump valves each with an upstream isolation valve, connects the high pressure turbine exhaust to the condensers. The system was designed to divert some steam away from the low pressure turbine in the event of a generator breaker opening, turbine trip or overspeed setpoint trip, thereby limiting the turbine-generator overspeed. This change eliminates the administrative requirement to arm the system during power escalation. System valves will be maintained closed, with power removed from the isolation valves.

At the presently-licensed power rating, which is less than the design power, the low pressure steam dump system is not required to satisfy turbine overspeed limits as described in the FSAR turbine missile analysis and can be isolated. The change does not involve an unreviewed safety question and the technical specifications and bases are unaffected.

61. Low Pressure Dump Motor Operated Stop Valve Bypass

The low pressure steam dump system consists of six dump valves, each in series with an upstream isolation valve. The system is designed to bypass steam from the high-pressure turbine exhaust directly to the condenser in the event of a turbine-generator trip, thereby limiting its overspeed. As originally installed, each isolation valve (located on the high pressure side of the dump valve) had a small bypass line from the inlet to the outlet of the valve. A trap was located between the isolation and dump valves to drain condensate to the condenser. With the isolation valve closed, condensate collected above the valve would flow through the bypass to the trap and then to the condenser. At times when dump valve leakage required the stop valve to be closed, a significant amount of steam flowed through the open bypass and then to the condenser through the dump valve. To stop this steam loss, each bypass line was rerouted from the inlet side of the isolation valve through a shutoff valve to the trap.

In the normal operating configuration (dump valve closed and isolation valve open) the system operates as before. With the isolation valve closed due to bypass valve leakage, condensate collected upstream of the isolation valve is routed to the trap but steam will not leak to the condenser.

There is no change to the technical specifications involved and there is no unreviewed safety question involved.

62. Improvements to Main Boiler Feed Pumps

Various mechanical changes were made to the instrumentation and hydraulic control systems of the main boiler feed pumps to improve reliability and reduce the number of feedwater transients. System functional requirements were not changed. The main boiler feed pumps are not safety-related and the control systems are not part of the technical specifications. This change, therefore, does not involve an unreviewed safety question or a change to the technical specification.

63. Reduction of Operator Actions on Main Boiler Feed Pump Trip

Immediately following loss of a main boiler feed pump, several operator actions have been required to attempt to prevent a unit trip. These have included running back the turbine load and starting any idle condensate pump. In most cases these actions did not prevent a unit trip because of the timing required. The unit has been modified to make these two actions occur automatically. Circuitry has been added to sense the speed of the main boiler feed pumps (MBFP), and on low MBFP speed, will automatically start an idle condensate pump and initiate turbine runback. A circuit has also been added to start any idle condensate pump on low pressure at the suction of the MBFP, which is indicative of a trip of a condensate pump. These changes are intended to assist the operator with actions now performed manually to help reduce the number of unit trips and challenges to the protection systems.

The Condensate System is not a safety-related system. Even without the automatic start capability, the plant can be operated with three condensate pumps. The automatic start of a condensate pump will not result in a change in feedwater enthalpy. The FSAR analysis of a steamline break with continued feedwater addition accident assumes all three condensate pumps are operating.

The initiation of a turbine runback can be considered a "partial loss of load". The FSAR includes a full loss of load analysis which assumes concurrent loss of all main feedwater. This is the bounding analysis for transients that include a complete or partial loss of load and remains unaffected by this change.

The technical specification and its bases are unaffected and these changes do not involve an unreviewed safety question.

64. Replace Condensate System Valve LCV-1158

Condensate System valve LCV-1158, in the line between the condensate tank outlet and the main condensers, was replaced. The replacement valve has been seismically analyzed, and the piping has been re-analyzed to take into account the greater weight of the new valve. The two valves are functionally equivalent. Therefore, no unreviewed safety question is involved.

65. Condensate Pump Seal Water Low Pressure Alarm Separation

A single alarm was previously used to indicate both "6.9 kV Motor Tripped" (which annunciates if a running pump motor trips) and "Common Condensate Pump Seal Water Low Pressure". Two separate alarms have now been provided for these conditions. The alarm systems and condensate pumps are not safety-related, and no unreviewed safety question is involved. This change does not affect the technical specification or its bases.

66. Auxiliary Boiler Feed Pump Start Circuit Modification

Circuitry changes were made to enhance the automatic starting reliability of the motor driven Auxiliary Boiler Feed Pumps. On actuation of the Safety Injection System, starting of the auxiliary boiler feed pumps for loss of feedwater and loss of offsite power is blocked to prevent improper load sequencing onto the diesel generator. Changes were made to prevent a single relay failure in the block circuitry from preventing the start of both Auxiliary Boiler Feed Pumps. These changes involved wiring to a second independent relay for AFW pump 23 and do not involve an unreviewed safety question.

67. Auxiliary Boiler Feed Flow Control

The control systems for the feedwater control valves associated with the motor-driven Auxiliary Boiler Feed Pumps have been changed. Originally, the positions of the valves were directly controlled by the plant operators, which allowed flow to vary as steam generator pressure changed. The modification was to install closed-loop flow control systems, in which valve position is automatically changed to maintain the preset flow. The operator maintains control over the system through the flow control setpoint. The existing pressure runout protection system remains in the system and will override the control system signal as required to maintain a minimum pump discharge pressure. The flow transmitters utilized in the system measure total flow to each steam generator. If a steam generator is being fed by both the steam-driven and the motor-driven auxiliary boiler feed pumps, the system will modulate the motor driven pump regulating valves to maintain the preset flow, within its capabilities. The flow controllers are capable of withstanding a safe shutdown earthquake and continue to maintain functional capability. The operator can still manually override the controller to change valve positions as was the case with the original controllers. This modification does not affect the bases for or ability to meet technical specifications and does not involve an unreviewed safety question.

68. Add Flow Control Valves to the Instrument Air Line for Replacement Valve PCV-1139

Valve PCV-1139 admits steam to the turbine which drives Auxiliary Boiler Feed Pump No. 22. Upon occurrence of an open signal, PCV-1139 may go to the demanded position with excessive speed, resulting in a pressure transient. To avert this problem, two adjustable valves were installed in the pneumatic line between the valve operator and its positioner. The valves are set to optimize the stroke time, setting it to less than the FSAR limit of one minute. The valves are lock-wired to assure their position is maintained. The weight of the valves is such that the existing seismic design is not affected. This installation will assure smooth, controlled opening of PCV-1139 and further reduce the potential for opening relief valve MS-52 or overspeeding the pump-turbine upon pump start. The addition of these valves does not affect the basis for any technical specification and does not involve an unreviewed safety question.

69. Replace Auxiliary Boiler Feed Pump No. 23 Discharge Pressure Transmitter PT-406B

The pressure transmitter at the discharge of Auxiliary Boiler Feed Pump No. 23 was replaced. The old and new transmitters are functionally the same, but the new one is heavier. A new analysis was performed which verified seismic acceptability. Therefore, no unreviewed safety question is involved. Because the transmitters are functionally the same, the ability to meet technical specification requirements is not affected.

70. Replace Auxiliary Boiler Feed Pump No. 22 Governor Control Arm

A plant-fabricated governor control arm anchor plate was installed on Auxiliary Boiler Feed Pump 22 to replace a broken one. The replacement plate is more substantial than the previous one, and calculations have verified its strength and seismic adequacy as a permanent replacement. No unreviewed safety questions are involved in this change.

71. Equivalent Replacement of BFD-77; Auxiliary Feedwater Condensate Storage Tank Recirculation Valve

Manual valves BFD-77 and BFD-78 are located in the recirculation lines from the motor-driven Auxiliary Boiler Feed Pumps to the Condensate Storage Tank. Valve BFD-77 was replaced with a valve of a different design which is heavier than the original valve. Accordingly, pipe supports were added so the piping will retain its integrity in the event of a safe shutdown earthquake. The valves parallel the flow control valves which assure minimum pump flow, and are closed during normal plant operation and whenever the auxiliary feedwater system is required to be operable. This change does not involve any technical specification or unreviewed safety question.

72. Auxiliary Reactor Cavity Pit Sump Pump Temporary Installation

A temporary air operated pump, with associated hoses for suction, discharge, and air supply, was installed in the reactor cavity pit. It provided leakage control during normal operation and served no safety function.

The installation was is such that the components could withstand a safe shutdown earthquake without becoming missiles. The containment isolation valves in the line which supplies air to the pump would be operated in accordance with the Technical Specifications. The pump contained uncoated aluminum which was assumed to corrode and generate hydrogen during a loss of coolant accident. The hydrogen concentration in the containment during this postulated event was assessed based on actual power level versus assumed FSAR levels, and shown to remain within the limits of the FSAR analysis and the bases of the technical specifications. Therefore, this change does not involve an unreviewed safety question or a change in the technical specification. This pump was removed during the Cycle 8/9 refueling outage.

73. Post-Accident RCS Liquid Sampling System - Installation of Boron Analyzer and LSP Sump Pumps

A post-accident sample panel was installed to satisfy the requirements of NUREG 0578. The modification consisted of the installation of a dissolved gas-analyzer and an ion chromatograph to provide additional analysis and monitoring capability. Changes were also made to the Post Accident Sampling System instrumentation, control valve and pumps. Relief valves 1301A and 1301B on pressurizer sample lines 25 and 26 in the primary sampling system were removed. The sample panel contains pressure reducing and pressure relief valves to perform the same function as the removed valves. The sample panel valves vent to the High Radiation Sampling System collection tank.

The changes affect the High Radiation Sampling System which provides information to plant operators. Its operation has no effect on existing safety-related systems.

No unreviewed safety question and no change to the technical specifications are involved by these changes to the facility.

74. Installation of a Second Sump Pump and Controls in the 15 Ft. El. Sump Pit of the Primary Auxiliary Building

A second sump pump was installed in the sump on the 15 ft. elevation in the primary auxiliary building. It serves as a backup to the existing sump pump, and will discharge to the Liquid Waste Holdup Tank. It does not involve an unreviewed safety question or a change to the technical specifications.

75. Modification to Sampling Lines for Post-Accident Radiation Monitoring System Monitor R-27

Post-accident radiation monitor R-27 is provided to monitor levels of noble gases in the plant vent under accident conditions. The sample line between the plant vent and the monitor was replaced with a larger line, and process instrumentation and valving were added. These changes do not affect the system functionally, and are intended to improve the reliability and flexibility of the system. The system and modification are safety-related Class A and Seismic Category I. This modification has not changed the requirements of the technical specifications and does not involve an unreviewed safety question.

76. Temporarily Defeat Malfunctioning Alarms on Iodine-131 Monitors

The iodine-131 monitors for the main station vent, blowdown area, containment purge and PAB/decay tank discharge have had low sample flow and loss of signal category alarm inputs disabled. High radiation alarm input signals (automatic functions) remain functional.

Process radiation monitors R-11 and R-13 monitor airborne particulate radioactivity in the containment and in the plant vent, respectively. Monitors R-14 and R-19 monitor radiation from the plant vent gas and steam generator liquid, respectively. In addition, other monitors exist to provide post accident monitoring. Since other monitors provide protection for and monitor plant operation and releases, and the capability to fulfill these protective functions is not affected, this jumper was determined not to involve an unreviewed safety question.

77. In-House Radio System

An in-house radio system has been added to provide the plant with instant two-way radio communications between fixed points and field personnel. The system uses low-wattage handheld units. It has no safety function and its failure would not affect any safety system. It does not involve an unreviewed safety question.

78. Radiation Sampling Program in Unit 1

As part of a research program on residual radioactive contamination in retired light water reactors, samples of piping, concrete and boiler scale were taken from Indian Point Unit 1. Samples were taken only from retired Unit 1 systems which do not provide support or safety related functions for Unit 2, and do not affect the ability of Unit 2 to achieve safe shutdown in the event of an accident. The sampling program has no effect on fire protection or the seismic design of either Unit 1 or Unit 2, and will involve no uncontrolled or unmonitored radioactive releases. The change has no relation to the Unit 2 FSAR. It does not involve an unreviewed safety question or a change to the technical specifications.

79. **Installation of Expansion Bolts, Anchor Bolts and Dowels in Primary Auxiliary Building to Support Passageway and Overhead Crane**

A passageway is being installed between the Maintenance and Outage Building and the Primary Auxiliary Building, and an overhead crane system is being installed between the Maintenance and Outage Building and the Containment Building Equipment Hatch. To support these structures, expansion bolts, anchor bolts, and dowels have been installed in the Waste Holdup Tank Pit Building, in the tunnel between the Primary Auxiliary Building and the Boric Acid Building, and in the containment building wall. Installation of the bolts will not affect the seismic or structural capabilities of any of the buildings, and the weight and maximum load supported will be well within the stress limits of the reinforced concrete. This change is not involved with any technical specification and does not involve an unreviewed safety question.

80. **Provide Openings in the Electrical Penetration Tunnel and the Primary Auxiliary Building Exhaust Plenum**

Two holes were bored in the electrical penetration tunnel and a section of 6 inch piping was installed and grouted in to each hole. Penetrations were also installed in the Primary Auxiliary Building exhaust duct. The pipe and exhaust duct penetrations are for use only during containment integrated leak rate testing. During normal plant operations the duct penetrations are closed with blind flanges. The electrical penetration pipes are filled with material to prevent radiation streaming and closed by blind flanges. The duct penetrations are Seismic Category I, the seismic integrity of the electrical penetration tunnel and walls is not affected, and the piping is seismically supported. Therefore, this modification does not involve an unreviewed safety question.

81. Generic Solenoid Valve Replacement

Exact replacements for certain solenoid valves are no longer available. Existing valves are being replaced on an "as-needed" basis. The following criteria are applied to replacements:

- The replacement will perform the same function as the original.
- The replacement will meet or exceed the specification requirements of the original.
- Seismic analyses or reviews will be performed when required by differences between the original and the replacement valve and/or installation.

Thus, the replacement valves will be functionally and physically equivalent to the original valves and the specific evaluation of such will be documented on a case-by-case basis. These changes, therefore, do not involve an unreviewed safety question.

82. Generic Replacement of EQ Limit Switches

To maintain operability, limit switches which have reached the end of their environmentally-qualified lifetimes were replaced. The replacement limit switches meet or exceed the design specification, applicable regulatory criteria and commitments made to the NRC. The reliability, availability, response times and other pertinent functional characteristics are at least equivalent to the original. These replacements do not involve an unreviewed safety question.

83. Generic Continental Manual Butterfly Valve Replacement

Exact replacements are not available for certain manually operated butterfly valves used in safety related systems. These valves are being replaced on an "as-needed" basis and are governed by the criteria that the replacement valve will perform the same function as and will meet or exceed the specification requirements of the original valve. If the weight of the replacement valve deviates from that of the original, or if it is not installed in the same orientation and position, seismic analysis or review will be performed and additional seismic supports will be installed as required. Fire protection impacts will also be reviewed. Satisfaction of appropriate criteria for specific valves is documented on a case-by-case basis. These changes do not involve an unreviewed safety question.

84. Generic Replacement of Copper Instrument Tubing with Stainless Steel Tubing

Copper instrument tubing used throughout the plant for both safety and non-safety related systems will be replaced with stainless steel tubing on an as-needed basis or when maintenance is performed on an existing copper system. Stainless steel tubing meets or exceeds the requirements of the copper tubing it is replacing as specified in the Con Edison specification "Installation of Tubing", and will follow the same routing as the tubing being replaced. All tubing will be seismically supported as required. This modification does not involve the basis of a technical specification or an unreviewed safety question.

85. Removal of No. 22 Boric Acid Evaporator, Gas Stripper and Ion Exchange Filter

No. 22 Boric Acid Evaporator, Gas Stripper and Ion Exchange Filter have been removed. These components were part of the Chemical and Volume Control System, and were originally provided for use in recovering and recycling boron which is no longer done. Liquids previously processed by this system are now routed to either the Volume Control Tank or to the Waste Disposal System. Connecting piping has been cut and capped, and additional supports added where needed to ensure the seismic qualification. Operation of the boric acid evaporator package is not assumed in any of the safety analyses. Removal of this equipment does not involve an unreviewed safety question.

86. Removal of Boron Injection Tank

The boron injection tank has been retired from service at Indian Point Unit 2 since 1985. The physical removal of the tank was implemented.

The mechanical portion of the disconnection was previously reported, and 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.

Accordingly, the physical removal process has been such that remaining systems were not adversely affected and will continue to perform their intended functions. This change did not involve an unreviewed safety question.