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December 5, 2000

Re: Indian Point Unit No. 2
Docket No. 50-247
NL 00-143

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop P1-137
Washington, D.C. 20555-0001

SUBJECT: Indian Point Unit 2 Technical Specification Bases Changes

Consolidated Edison Company of New York, Inc. hereby submits changes that have been made to the Indian Point Unit 2 Technical Specification (TS) Bases. These changes have been processed in accordance with the provisions of 10 CFR 50.59.

Description of Changes

The Bases for TS 2.3, "Limiting Safety System Settings, Protective Instrumentation," have been revised to remove the reference to Updated Final Safety Analysis Report (UFSAR) Table 7.4-2. This table is being deleted from the UFSAR.

The Bases for TS 3.3, "Engineered Safety Features," have been revised to reflect changes made to TS 3.3 in TS Amendment No. 163 (June 7, 1993). The comparable changes to the TS Bases had been inadvertently omitted from the associated TS change request.

The Bases for TS 3.3 associated with the Isolation Valve Seal Water System (IVSWS) are being changed to provide additional clarification on the operation of the system. First, the description of the IVSWS regarding containment design pressure is deleted. As described in the UFSAR, this is not a function of the IVSWS. Secondly, additional clarification is provided to describe the flow paths

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necessary to consider IVSWS operable. This change is consistent with the design of the IVSWS.

The Bases for TS 3.6, "Containment System," have been revised to delete an incorrect statement regarding containment internal pressure limitations.

The Bases for TS 4.5, "Engineered Safety Features," associated with the passive autocatalytic recombiners (PARs) is being revised to reflect the vendor's recommended hydrogen gas mixture. This change will allow the use of hydrogen gas mixtures recommended by the vendor.

The attachment provides the replacement pages for the TS Bases. There are no commitments contained in this submittal. Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing at (914) 734-5074.

Very truly yours,

Attachment

cc:

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ATTACHMENT TO NL 00-143

REVISED TECHNICAL SPECIFICATION BASES PAGES

Affected pages:

2.3-5
2.3-7
3.3-11
3.3-17
3.6-3
4.5-9

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performance within the design characteristics of the instruments through channel calibrations, drift evaluations, instrument response characteristics, and other manufacturer recommended tests.

Process rack modules or a sensor/transmitter found outside the "as left" band for calibration accuracy must be returned to within the band after the performance of each surveillance test.

The high flux reactor trips provide redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis[1].

The power range nuclear flux reactor trip high setpoint protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis[2].

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (UFSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is backed up by the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss of coolant accident. Its setting limit is consistent with the value assumed in the loss of coolant analysis[4].

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detector (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 60% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below the safety limit DNBRs during normal operational transients.

A turbine trip causes a direct reactor trip, when operating at or above 35% power, in order to reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of this trip. Functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

The steam-feedwater flow mismatch trip does not appear in the specification as this setting is not used in the transient and accident analysis (UFSAR Section 14).

To avoid mechanical interference due to thermal contraction between the fuel and the control rods, an automatic backup to manual tripping of the control rods is provided. Prior to T_{cold} decreasing below 381°F during RCS cooldown, the Control Rod Protection System will open the reactor trip breakers which unlatches the control rod drive shafts from the CRDMs.

References

(1) UFSAR 14.1.1
(2) UFSAR 14.1.2
(3) Deleted
(4) UFSAR 14.3.1
(5) UFSAR 14.1.2
(6) UFSAR 7.2
(7) UFSAR 3.2.1
(8) UFSAR 14.1.6
(9) UFSAR 14.1.9

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant[1]. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and therefore the minimum required engineered safeguards and auxiliary cooling systems are required to be operable. During low-temperature physics tests there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safeguards systems are not required.

When the reactor is critical, the probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus operation with the reactor critical with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. Inoperability of a single component does not negate the ability of the system to perform its function[2], but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. If (1) the inoperable component is not repaired within the specified allowable time period, or (2) a second component in the same or related system is inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The plant operating procedures require immediate action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. The specified repair times do not apply to regularly scheduled maintenance of the engineered safeguards systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on two considerations:

The seven-day out-of-service period for the Weld Channel and Penetration Pressurization System and the Isolation Valve Seal Water System is allowed because no credit has been taken for operation of these systems in the calculation of offsite accident doses should an accident occur. No other safeguards systems are dependent on operation of these systems[11]. The minimum pressure settings for the IVSWS and WC & PPS during operation assures effective performance of these systems. Portions of the Weld Channel Pressurization System are in areas that are not accessible, such as below the concrete floor of containment or in high radiation areas. If it is determined that it is not practicable to repair an inoperable portion of the system, then that portion may be disconnected.

The IVSWS seal water tank pressure is maintained by a nitrogen supply piped to the tank via piping that expands to three flow paths. Two flow paths contain a pressure control valve and the third path is a manual bypass around one of the control valves. Based on original plant design, two flow paths to the tank are required to consider IVSWS operable. If only one flow path to the tank is available, the appropriate LCO would be entered.

<u>References</u>

(1) UFSAR Section 9

(2) UFSAR Section 6.2

(3) DELETED

(4) UFSAR Section 6.4

(5) Reference Deleted

(6) UFSAR Section 9.3

(7) UFSAR Section 9.3

(8) UFSAR Section 9.6.1

(9) UFSAR Section 14.3

(10) Indian Point Unit No. 2, UFSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

(11) UFSAR Sections 6.5 and 6.6

(12) WCAP-12312, "Safety Evaluation for An Ultimate Heat Sink Temperature to 95°F at Indian Point Unit 2", July, 1989.

(c) Isolate each affected penetration within 4 hours by use of at least one closed manual valve[3)] or blind flange that meets the design criteria for an isolation valve, or

(d) Be in cold shutdown within the following 36 hours, utilizing normal operating procedures.

4. Non-automatic containment isolation valves may be added to plant systems, without prior license amendment to Table 3.6-1, provided that a revision to this Table is included in a subsequent license amendment application.

B. INTERNAL PRESSURE

If the internal pressure exceeds 2 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shut down.

C. CONTAINMENT TEMPERATURE

The reactor shall not be taken above the cold shutdown condition unless the containment ambient temperature is greater than 50°F.

Basis

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if a Reactor Coolant System rupture were to occur.

The shutdown margins are selected based on the type of activities that are being carried out. The shutdown margin requirement of Specification 3.8.B.2 when the head is off precludes criticality during refueling. When the reactor head is not to be removed, the specified cold shutdown margin of 1% $\Delta k/k$ precludes criticality at cold shutdown conditions.

The containment can withstand an internal vacuum of 2.5 psig. The 2.0 psig vacuum specified as an operating limit avoids any difficulties with motor cooling.

The hydrogen recombiner system is an engineered safety feature which would function following a loss-of-coolant accident to control the hydrogen evolved in the containment. The passive autocatalytic recombiners(PARs) contain no control or support equipment which would require surveillance. No specific degradation mechanism has yet been identified for the catalysts plates in standby service. Periodic visual examination and cleaning if necessary is done to prevent significant gas blockage by dust or debris. Representative plates are periodically removed and their response to an approximately 1.5% hydrogen gas mixture is evaluated for evidence of unexpected degradation.

The biannual testing of the containment atmosphere sampling system will demonstrate the availability of this system.

The recirculation fluid pH control system is a passive safeguard with the baskets of trisodium phosphate located in the containment sump area. Periodic visual inspections are required (Refueling#) to verify the storage baskets are in place, have maintained their integrity, and filled with trisodium phosphate.

The control room air filtration system is designed to filter the control room atmosphere for intake air during control room isolation conditions. The control room air filtration system is designed to automatically start upon control room isolation. High-efficiency particulate absolute (HEPA) filters are installed upstream of the charcoal adsorbers to prevent clogging of these adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radioiodine by control room personnel. The required in-place testing and the laboratory charcoal sample testing of the HEPA filters and charcoal adsorbers will provide assurance that Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50 continues to be met.

The fuel storage building air filtration system is designed to filter the discharge of the fuel storage building atmosphere to the plant vent. This air filtration system is designed to start automatically upon a high radiation signal. Upon initiation, isolation dampers in the ventilation system are designed to close to redirect air flow through the air treatment system. HEPA filters and charcoal adsorbers are installed to reduce potential releases of radioactive material to the atmosphere. Nevertheless, as required by Specification 3.8.B.6, the fuel storage building air filtration system must be operating whenever spent fuel is being moved unless the spent fuel has had a continuous 35-day decay period. The required in-place testing and the laboratory charcoal sample testing of the HEPA filters and charcoal adsorbers will provide added assurance that the criteria of 10 CFR 100 continue to be met.