Correction to Amdt 101 to DPR-26

Docket No. 50-247

December 5, 1985

DISTRIBUTION

Docket File	NRC PDR
LPDR	ORB#1 Rdg
Gray File	H. Thompson
C. Parrish	M. Slosson
OELD	L. Harmon
E. Jordan	B. Grimes
J. Partlow	T. Barnhart 4
W. Jones	M. Virgilio
ACRS 10	OPA, C. Miles
R. Diggs	R. Ballard

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

SUBJECT: AMENDMENT NO. 101 TO THE INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

By letter dated October 23, 1985, Amendment No. 101 to Facility Operating License DPR-26 for the Indian Point Nuclear Generating Unit No. 2 was issued. The amendment was issued with several pages containing administrative errors.

Page 3.1.A-3 was issued without incorporating the change made by Amendment No. 93.

Pages 3.1F-1, F-4 and F-5 were issued without incorporating the revisions contained in Amendment No. 85.

Pages 4.3-1 and 4.3-2 were issued without incorporating the revisions made by Amendment No. 96.

Page 4.9-1 was issued without the changes made by Amendment No. 94.

Pages 5.3-1 and 5.3-2 were issued without the revisions made by Amendment No. 100.

A copy of the corrected pages is enclosed.

In addition the Safety Evaluation was issued with several administrative errors. Revision 1 to the Safety Evaluation is enclosed.

Sincerely,

/s/S. Varga

Steven A. Varga, Director
PWR Project Directorate #3
Division of PWR Licensing-A

Enclosure:
As stated

8512130337 851205
PDR ADOCK 05000247
P PDR

cc:
See next page

*SEE PREVIOUS WHITE FOR CONCURRENCE

ORB#1:DL	ORB#1:DL	MMS	~~BC-ORB#1~~:DL
CParrish	MSlosson;ps		SVarga
11/12/85	12/3/85		12/5/85

Docket No. 50-247

DISTRIBUTION

Docket File	NRC PDR
LPDR	ORB#1 Rdg
Gray File	H. Thompson
C. Parrish	M. Slosson
OELD	L. Harmon
E. Jordan	B. Grimes
J. Partlow	T. Barnhart 4
W. Jones	M. Virgilio
ACRS 10	OPA, C. Miles
R. Diggs	R. Ballard

Mr. John D. O'Toole
Vice President
Nuclear Engineering and Quality Assurance
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. O'Toole:

SUBJECT: AMENDMENT NO. 101 TO THE INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

By letter dated October 23, 1985, Amendment No. 101 to Facility Operating License DPR-26 for the Indian Point Nuclear Generating Unit No. 2 was issued. The amendment was issued with several pages containing administrative errors.

Page 3.1.A-3 was issued without incorporating the change made by Amendment No. 93.

Pages 3.1F-1, F-4 and F-5 were issued without incorporating the revisions contained in Amendment No. 85.

Pages 4.3-1 and 4.3-2 were issued without incorporating the revisions made by Amendment No. 96.

Page 4.9-1 was issued without the changes made by Amendment No. 94.

Pages 5.3-1 and 5.3-2 were issued without the revisions made by Amendment No. 100.

A copy of the corrected pages is enclosed.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

cc:
See next page

ORB#1:DL CP	ORB#1:DL MMS	BC-ORB#1:DL
CParrish	MSlosson;ps	SVarga
11/12/85	11/12/85	11/ /85

Mr. John D. O'Toole
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station 1/2

cc:
Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Michael Blatt
Director Regulatory Affairs
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York, 10511

Robert L. Spring
Nuclear Licensing Engineer
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Carl R. D'Alvia, Esquire
Attorney for the Village of
Buchanan, New York
395 South Riverside Avenue
Croton-on-Hudson, New York 10520

Mr. Jay Dunkleberger
Office of Policy Analysis
and Planning
York State Energy Office
Building 2, Empire State Plaza
Albany, New York 12223

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Mr. Peter Kokolakis, Director
Nuclear Licensing
New York Power Authority
123 Main Street
White Plains, New York 10601

Mr. Murray Selman
Vice President, Nuclear Power
Consolidated Edison Company of
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Mr. Frank Matra
Resident Construction Manager
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenues
Buchanan, New York 10511

Ezra I. Bialik
Assistant Attorney General
Environmental Protection Bureau
New York State Department of Law
2 World Trade Center
New York, New York 10047

c. the event either a PORV(s) or CS vent(s) is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Nuclear Regulatory Commission within 30 days pursuant to Specification 6.9.2.h. The report shall describe the circumstances initiating the transient, the effect of the PORV(s) or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

5. Power Operated Relief Valves (PORVs)/Block Valves (for operation above 350°F)

a. Whenever the reactor coolant system is above 350°F, the PORVs and their associated block valves shall be operable with the block valves either open or closed.

b. If a PORV becomes inoperable when above 350°F, its associated block valve shall be maintained in the closed position.

c. If a PORV block valve becomes inoperable when above 350°F, the block valve shall be closed and deenergized.

d. If the requirements of specification 3.1.A.5.a, 3.1.A.5.b or 3.1.A.5.c above cannot be satisfied, compliance shall be established within four (4) hours, or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled below 350°F.

e. With regard to the use of the PORVs/Block Valves as a reactor coolant system vent, the requirements of specification 3.16 shall be adhered to.

6. Pressurizer Heaters

a. Whenever the reactor coolant system is above 350°F, the pressurizer shall be operable with at least 150kw of pressurizer heaters.

b. If the requirements of specification 3.1.A.6.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or the reactor shall be placed in the hot shutdown within the next six(6) hours and subsequently cooled below 350°F.

Basis

When the boron concentration of the Reactor Coolant System (RCS) is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. The requirement for at least one reactor coolant pump or one residual heat removal pump to be in operation is to provide flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. Below 350°F, a single reactor coolant loop or RHR loop provides sufficient heat removal

8512130344 851205
PDR ADOCK 05000247
P PDR

3.1.F. REACTOR COOLANT SYSTEM LEAKAGE AND LEAKAGE INTO THE CONTAINMENT FREE VOLUME

Specification

1. LEAKAGE DETECTION AND REMOVAL SYSTEMS

a. The reactor shall not be brought above cold shutdown unless the following leakage detection and removal systems are operable:

(1) Two containment sump pumps.
(2) Two containment sump level monitors.
(3) A containment sump discharge line flow monitoring system.
(4) Two recirculation sump level monitors.
(5) The reactor cavity continuous level monitoring system and an independent reactor cavity level alarm.
(6) Two of the following three systems:
(a) A containment atmosphere gaseous radioactivity monitoring system.
(b) A containment atmosphere particulate radioactivity monitoring system.
(c) The containment fan cooler condensate flow monitoring system.

b. When the reactor is above cold shutdown, the requirements of specification 3.1.F.1.a. may be modified as follows:

(1) One containment sump pump may be inoperable for a period not to exceed seven (7) consecutive days provided that on a daily basis the other containment sump pump is started and discharge flow is verified.

(2) One of the two required containment sump level monitors may be inoperable for a period not to exceed seven (7) consecutive days.

(3) The containment sump discharge line flow monitoring system may be inoperable for a period not to exceed seven (7) consecutive days provided a detailed Waste Holdup Tank water inventory balance is performed daily.

(4) One of the two required recirculation sump level monitors may be inoperable for a period not to exceed fourteen (14) consecutive days.

(5) One of the two required reactor cavity level monitors may be inoperable for a period not to exceed thirty (30) consecutive days.

(3) If water level in the containment sump reaches EL. 45' or the water level in the recirculation sump reaches EL. 35', or the water level in the reactor cavity reaches EL. 20', the reactor shall be placed in a cold shutdown condition within the next 36 hours unless the water level(s) is reduced below the specified limit(s).

(4) If the water level in the containment sump increases above EL. 45' and the water level in the recirculation sump increases above EL. 39'-9", or the water level in the reactor cavity increases above EL. 20' 5", immediately place the reactor in a subcritical condition and initiate an expeditious cooldown of the reactor to the cold shutdown condition.

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up soon as practicable.

Although some leak rates on the order of gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any pressure boundary of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are 0.025 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in reactor coolant system leakage of 1 gpm is detectable within 1 minute after it occurs.

b. The containment radiogas monitor is less sensitive than the air particulate monitor. The sensitivity range of the instrument is 10^{-2} μc/cc to 10^{-7} μc/cc.

c. In the event either a PORV(s) or a RCS vent(s) is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Nuclear Regulatory Commission within 30 days pursuant to Specification 6.9.2.h. The report shall describe the circumstances initiating the transient, the effect of the PORV(s) or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

5. Power Operated Relief Valves (PORVs)/Block Valves (for operation above 350°F)

a. Whenever the reactor coolant system is above 350°F, the PORVs and their associated block valves shall be operable with the block valves either open or closed.

b. If a PORV becomes inoperable when above 350°F, its associated block valve shall be maintained in the closed position.

c. If a PORV block valve becomes inoperable when above 350°F, the block valve shall be closed and deenergized.

d. If the requirements of specification 3.1.A.5.a, 3.1.A.5.b or 3.1.A.5.c above cannot be satisfied, compliance shall be established within four (4) hours, or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled below 350°F.

e. With regard to the use of the PORVs/Block Valves as a reactor coolant system vent, the requirements of specification 3.16 shall be adhered to.

6. Pressurizer Heaters

a. Whenever the reactor coolant system is above 350°F, the pressurizer shall be operable with at least 150kw of pressurizer heaters.

b. If the requirements of specification 3.1.A.6.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or the reactor shall be placed in the hot shutdown within the next six(6) hours and subsequently cooled below 350°F.

Basis

When the boron concentration of the Reactor Coolant System (RCS) is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. The requirement for at least one reactor coolant pump or one residual heat removal pump to be in operation is to provide flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. Below 350°F, a single reactor coolant loop or RHR loop provides sufficient heat removal

(3) If water level in the containment sump reaches EL. 45' or the water level in the recirculation sump reaches EL. 35', or the water level in the reactor cavity reaches EL. 20', the reactor shall be placed in a cold shutdown condition within the next 36 hours unless the water level(s) is reduced below the specified limit(s).

(4) If the water level in the containment sump increases above EL. 45' and the water level in the recirculation sump increases above EL. 39'-9", or the water level in the reactor cavity increases above EL. 20' 5", immediately place the reactor in a subcritical condition and initiate an expeditious cooldown of the reactor to the cold shutdown condition.

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up soon as practicable.

Although some leak rates on the order of gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any pressure boundary of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are 0.025 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in reactor coolant system leakage of 1 gpm is detectable within 1 minute after it occurs.

b. The containment radiogas monitor is less sensitive than the air particulate monitor. The sensitivity range of the instrument is 10^{-2} μc/cc to 10^{-7} μc/cc.

c. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units including leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows from approximately 1 gpm to 15 gpm per detector can be measured by this system. Condensate flows greater than 15 gpm can be determined using weir calibration curves. Condensate flows less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.

d. Leakage detection via the containment sump level and discharge flow monitoring systems will determine leakage losses from all fluid systems to the containment free volume. Water collecting on the containment floor will normally be delivered to the containment sump via the containment floor trench system. Level monitoring of the containment sump is in part provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certian discrete levels within the containment sump. In addition, another level monitoring device provides a continuous level readout in the control room. When the water level in the containment sump reaches predetermined levels, one or both containment sump pumps will automatically start and pump the fluid out of containment to the liquid waste disposal system. Flow in the containment sump pump discharge line from containment to the Waste Holdup Tank is monitored on a continuous basis. Thus, monitoring of both the flow indication systems will provide a positive means for determining leakage into the containment free volume.

e. Water may also collect in the recirculation sump and/or the reactor cavity depending on the size and location of the leak. However, under most circumstances, the containment sump will be filled prior to the recirculation sump filling and both sumps will be filled prior to water level increasing on containment floor (EL. 46') sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. In addition, another level monitoring device provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a single analog continuous level indication in the control room and two separate and independent level switches each of which actuates an audible alarm in the control room.

4.3 REACTOR COOLANT SYSTEM INTEGRITY TESTING

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To specify tests for Reactor Coolant System integrity after the system is closed following normal opening, modification or repair.

Specification

a) When the Reactor Coolant System is closed after it has been opened, the system will be leak tested at not less than 2335 psig at NDT requirements for temperature.

b) When Reactor Coolant System modification or repairs have been made which involve new strength welds on components, the new welds shall meet the requirements of the applicable version of ASME Section XI as specified in the Con Edison Inservice Inspection and Testing Program in effect at the time.

c) The Reactor Coolant System leak test temperature-pressure relationship shall be in accordance with the limits of Figure 4.3-1 for heatup for the first fifteen (15) effective full-power yrs. of operation. Figure 4.3-1 will be recalculated periodically. Allowable pressure during cooldown for the leak test temperature shall be in accordance with Figure 3.1.B-2.

Basis

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system doe not leak at 2335 psig (Operating pressure + 100 psi: ± 100 psi is normal system pressure fluctuation), it will be leak tight during normal operation.

For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak tightness during normal operaton.

The inservice leak temperatures are shown on Figure 4.3-1. The temperatures are calculated in accordance with ASME Code Section III, 1974 Edition, Appendix G. This Code requires that a safety factor of 1.5 times the stress intensity factor caused by pressure be applied to the calculation.

For the first fifteen (15) effective full-power years, it is predicted that the highest RT_{NDT} in the core region taken at the 1/4 thickness will be 176°F. The minimum inservice leak test temperature requirements for periods up to fifteen (15) effective full-power years are shown on Figure 4.3-1.

The heatup limits specified on the heatup curve, Figure 4.3-1, must not be exceeded while the reactor coolant is being heated to the inservice leak test temperature. For cooldown from the leak test temperature, the limitations of Figure 3.1.B-2 must not be exceeded. Figures 4.3-1 and 3.1.B-2 are recalculated periodically, using methods discussed in WCAP-7924A and results of surveillance specimen testing, as covered in WCAP-7323.

Reference

1. FSAR, Section 4

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

Following a normalization of the computed boron concentration as a function of burn-up, actual boron concentration of the coolant shall be periodically compared with the predicted value.

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity is compared with that predicted. This process of normalization shall be completed early in core life. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive rod in the fully withdrawn position is always maintained.

Applicability

Applies to the reactor core, reactor coolant system, and emergency core cooling systems.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods.(1)

2. Deleted

3. The enrichment of reload fuel will be no more than 3.5 weight per cent U-235.

4. Deleted

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.(5)

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.(6) Design values for system temperature and pressure are 650°F and 2485 psig, respectively.

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in he vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,350 cubic feet.

References

(1) FSAR Section 3.2
(2) Deleted
(3) Deleted
(4) Deleted
(5) FSAR Sections 3.2
(6) FSAR Table 4.1-9

REVISION 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

DECAY HEAT REMOVAL AND ROD BANK WITHDRAWAL

Introduction

A number of events have occurred at operating Pressurized Water Reactor (PWR) facilities where decay heat removal capability has been seriously degraded due to inadequate administrative controls during shutdown modes of operation. One of these events, described in IE Information Notice 80-20 occurred at the Davis-Besse Station, Unit No. 1, on April 19, 1980. In IE Bulletin 80-22 dated May 9, 1980, licensees were requested to immediately implement administrative controls which would ensure that proper means are available to provide redundant methods of decay heat removal. While the function of the bulletin was to effect immediate action with regard to this problem, the NRC considered it necessary that an amendment be made to each PWR license to provide for permanent long-term assurance that redundancy in decay heat removal capability will be maintained. By the letter dated June 11, 1980, all PWR licensees were requested to propose Technical Specification changes that provide for redundancy in decay heat removal capability in all modes of operation; to use the NRC model technical specifications to provide an acceptable solution to the concern; to include an appropriate safety analysis as a basis; and to submit the proposed Technical Specification changes along with the bases by October 11, 1980.

Consolidated Edison Company of the New York (Consolidated Edison) responded to the NRC request for amended Technical Specifications for the Indian Point Nuclear Generating Unit 2, with their letter dated February 14, 1983, and proposed revisions to the Technical Specifications for the plant. In a letter dated September 14, 1984, Consolidated Edison responded to NRC staff's request for additional information and indicated that the existing and previously proposed Technical Specifications, supplemented by the surveillance requirements contained in the operational procedures presently available, satisfied the NRC requirements of assuring redundant decay heat removal for all modes of reactor operation.

By letter dated February 28, 1985, Consolidated Edison submitted a supplemental application revising the Technical Specifications to change the number of reactor coolant pumps required to be operating while control

8512130408 851205
PDR ADOCK 05000247
P PDR

rods were being withdrawn in operating Mode 3. This revision is being made to be consistent with Consolidated Edison's reanalysis of the Uncontrolled Rod Cluster Assembly (RCCA) bank withdrawal event initiated from a just critical hot zero power condition.

Discussion and Evaluation

The proposed technical specifications provide for redundant means of decay heat removal in all modes of operation except during refueling when a large mass of water is above the core. These redundant means are outlined as follows:

Power Operation and Startup

All reactor coolant loops must be in operation with all four reactor coolant pumps operating. However, operation with three reactor coolant pumps is permitted if the nuclear overpower trip setpoint is appropriately reduced.

Hot Standby

All reactor coolant loops are required to be operable. However, only two of the loops is required to be in operation.

Hot Shutdown

At least two of the heat removal coolant loops are required to be operable, (i.e., either two RC loops or two DHR loops, or a combination of one each of these coolant loops). One of the loops must be in operation.

Cold Shutdown and Refueling with the Water Level Above the Core Less Than 23 Feet

Two decay heat removal loops are required to be operable with at least one loop in operation.

Refueling with the Water Level Above the Core Greater Than 23 Feet

At least one decay heat removal loop is required to be in operation. The other loop need not be operable.

In addition to the above requirements for operability, the Technical Specification revisions specify surveillance intervals for heat removal systems that are consistent with the Standard Technical Specifications. The revised Technical Specifications provide an improvement over the existing ones since redundant decay heat removal will now be provided in MODES 3, 4, and 5. In MODE 6, with a large mass of water above the core, only a single heat removal path is required. The surveillance requirements that would identify any inoperable equipment or degrading performance are performed during each shift. The staff therefore concludes that the proposed Technical Specifications meet the intent of the Standard Technical Specifications with respect to redundant means of decay heat removal capability.

Additionally, Consolidated Edison has reanalyzed the uncontrolled RCCA event initiated from a just critical hot zero power condition with the assumption of two reactor coolant pumps in operation. The staff has examined the methods used to determine the peak fuel rod response and the input to that analysis, such as reactivity insertion rate, moderator on fuel temperature feedback effects, initial power distributions and reactor trip assumptions. The results of the analysis show that the event is terminated by the power range high neutron flux trip low setting of 35% of nominal power. The minimum Departure from Nucleate Boiling Ratio (DNBR) remains above the 1.30 limit at all times during the transient, thus assuring that the core and the reactor coolant systems are not adversely affected. Since the fuel rods do not experience DNB, there is no clad damage and, therefore, there are no radiological consequences associated with the event. Therefore, the staff concludes that the proposed Technical Specifications are acceptable.

OVERPRESSURE PROTECTION

Introduction

By letter dated August 13, 1976, the NRC requested Consolidated Edison to propose system modifications to provide overpressure protection for the reactor vessel. As discussed in our June 22, 1984 Safety Evaluation, Consolidated Edison has installed an acceptable overpressure protection system (OPS). By letter dated February 14, 1983, Consolidated Edison proposed Technical Specification changes to incorporate limiting conditions of operation and surveillance requirements associated with the overpressure protection system by the addition of new specifications that define the low temperature reactor coolant overpressure protection range, incorporate a limit on the maximum primary-to-secondary differential temperature that is permitted prior to starting a reactor coolant pump and incorporates new requirements on the operability of Power Operated Relief Valves (PORV's). Consolidated Edison also proposed changes to limit the maximum number of energized safety injection and/or charging pumps for various pressurizer pressure, pressurizer level, and temperature conditions in the reactor coolant system.

By letter dated February 28, 1985 Consolidated Edison submitted a supplement to the February 14, 1983 submittal which revised the proposed overpressure protection Technical Specifications. The revisions were required due to changes in the pressure-temperature limitations for heatup and cooldown which were also proposed in the February 28, 1985 submittal and were issued by Amendment No. 96. By letters dated August 1, 1985 and September 13, 1985 Consolidated Edison provided additional page changes which were inadvertently omitted from the February 28 submittal. These changes were administrative in nature and in no way changed the meaning or technical content of the previous submittals.

Consolidated Edison's February 14, 1983 application for amendment also proposed editorial and format changes to the Technical Specifications.

Discussion and Evaluation

By letter dated February 14, 1983, Consolidated Edison submitted proposed Technical Specifications for the OPS. The Technical Specification states that the OPS is required to be operable when the RCS temperature is less than or equal to 295°F, and it gives conditions on the operation and operability of the system. The proposed basis for this Technical Specification is as follows:

> The Overpressure Protection System is designed to relieve the RCS pressure for certain unlikely overpressure transients to prevent these incidents from causing a peak RCS pressure exceeding 10 CFR 50, Appendix G limits. When the OPS is "armed", MOVs 535 and 536 are in the open position, and the PORVs will open upon receipt of the appropriate signal. This OPS arming can be accomplished either automatically by the OPS when the RCS is below a prescribed temperature or manually by the operator.
>
> The OPS will be set to cause the PORVs to open at a pressure sufficiently low to prevent exceeding the Appendix G limits for the following events:
>
> 1. Startup of a reactor coolant pump with no other reactor coolant pumps running and the steam generator secondary side water temperature higher than the RCS water temperature.
> 2. Letdown isolation with three charging pumps operating.
> 3. Startup of one safety injection pump.
> 4. Loss of residual heat removal causing pressure rise from heat additions from core decay heat or reactor coolant pump heat.
> 5. Inadvertent activation of the pressurizer heaters.
>
> Consideration of the above events provides bounding PORV setpoints for other potential overpressure conditions caused by heat or mass additions at low temperature.
>
> The RCS is protected against overpressure transients when RCS temperature is less than or equal to 295°F by: (1) restricting the number of charging and safety injection pumps that can be energized to that which can be accommodated by the PORV's or the gas space in the pressurizer, (2) providing administrative controls on starting of a reactor coolant pump when the primary water temperature is less than the secondary water temperature, or (3) providing vent area from the RCS to containment for those situations where neither the PORV's nor the available pressurizer gas space are sufficient to preclude the pressure resulting from postulated transients from exceeding the limits of 10 CFR 50, Appendix G.
>
> The restrictions on starting a reactor coolant pump with the secondary side water temperature higher than the primary side will prevent RCS overpressurizations from the resultant volumetric swell into the

pressurizer that is caused by potential heat additions from the startup of a reactor coolant pump without any other reactor coolant pumps operating. When pressurizer level is between 30 and 85% of span, protection is provided through the use of the PORV's. When pressurizer level is less than 30% of span additional restrictions on pressurizer pressure make reliance on the PORV's unnecessary since the gas compression resulting from the insurge of liquid from the RCS pump start is insufficient to cause RCS pressure to exceed the Appendix G limits. The same method, i.e., control of pressurizer pressure and level, is used to accommodate the mass insurge into the pressurizer from safety injection and charging pump starts when the PORV's are not operational.

An additional restriction is put on the reactor coolant pump start when the secondary system water temperature is less than or equal to 40°F higher than the primary system water temperature and the pressurizer level is greater than 30%. This restriction is to prohibit starting the first reactor coolant pump when the RCS temperature is between 267°F and 295°F. The purpose of the restriction is to assure that the temperature rise resulting from the transient will not be outside limits for OPS actuation.

When comparison to the Appendix G limits is made, the comparison is to the isothermal Appendix G curve. Other than the delay time associated with opening the PORVs, and the error caused by non-uniform RCS metal and water temperatures during heat addition transients, the analysis does not make any allowance for instrument error. Instrument error will be taken into account when the OPS is set; i.e., the instrumentation will be set so that the PORVs will open at less than the required setpoint including allowance for instrument errors.

The determination of reactor coolant temperature may be made from the Control Room instrumentation. The determination of the steam generator water temperature may be made in the following ways:

(a) Assuming that the secondary side water temperature is at the saturation temperature corresponding to the secondary side steam pressure indicated on the Control Room instrumentation, or

(b) Conservatively assuming that the secondary side water temperature is at the reactor coolant temperature at which the last RCP was stopped during cooldown, or

(c) Actual or inferred measurement of the secondary side steam generator water temperature at those times it can be measured (such as return from a refueling outage).

The proposed technical specifications also contain surveillance requirements on the OPS. Specifically system testing requirements are included in Table 4.1-1 (Item 37) of the February 14 submittal. Valve operability and cycling tests for PORV's and MOV's associated with the OPS

are performed in accordance with the requirements of 10 CFR 50.55a. This is identified in the basis for Specification 4.18. The staff concludes that with the inclusion of the Technical Specification revisions the Indian Point 2 overpressure protection system meets GDC 15 and 31 and the Consolidated Edison has implemented the guidelines of NUREG-0224. Therefore, the Technical Specification changes are acceptable.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 23, 1985

Principal Contributors:

E. Lantz
V. Leung
M. Slosson