Attachment A

IP-2 Technical Specifications
Proposed Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 23, 1988

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

Specifications

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) two reactor cavity level monitors,

(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.

(6) Two of the three monitoring systems specified in Specification 3.1.F.1.a.(6) may be inoperable for a period not to exceed thirty (30) consecutive days. If either of the radioactivity monitoring systems specified in Specification 3.1.F.1.a.(6) is inoperable, grab samples of the containment atmosphere shall be obtained and analyzed daily.

c. If the conditions of Specification 3.1.F.1.b cannot be met or an inoperable system(s) is not restored to operable status within the time period(s) specified therein, then either perform a visual inspection of containment once a shift, or place the reactor in the hot shutdown condition within the next 6 hours and, if the inoperability continues, place the reactor in the cold shutdown condition within the following 30 hours.

2. Operational Leakage Limits

a. Primary to Secondary Leakage

(1) Primary to secondary leakage through the steam generator tubes shall not exceed 0.3 gpm in any steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours.

(2) If leakage from two or more steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24

hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If two steam generator tube leaks attributable to the tube denting phenomena are observed after the reactor is in cold shutdown, Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Whenever the reactor is shut down in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the NRC shall be informed before any tube is plugged or, if no tube is plugged, before the steam generator is returned to service.

b. RCS/RHR Pressure Isolation Valves Leakage

(1) Whenever the reactor is above cold shutdown, leakage through each of the RCS/RHR pressure isolation valves 897A, B, C and D, and 838A, B, C and D shall satisfy the following acceptance criteria:

(a) Leakage rates of less than or equal to 1.0 gpm are acceptable.

(b) Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

(c) Leakage rates greater than 5.0 gpm are unacceptable.

(2) If any RCS/RHR pressure isolation valve listed in Specification 3.1.F.2.b.(1) is determined to be inoperable based on the acceptance criteria presented therein, an orderly plant shutdown shall be initiated and the reactor shall be placed in the cold shutdown condition within 24 hours.

c. Total Reactor Coolant System Leakage

(1) Whenever the reactor is above cold shutdown, reactor coolant system leakage shall be limited to:

(a) No PRESSURE BOUNDARY LEAKAGE,

(b) 1 gpm UNIDENTIFIED LEAKAGE, and

(c) 10 gpm IDENTIFIED LEAKAGE.

(2) With any PRESSURE BOUNDARY LEAKAGE, the reactor must be placed in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

(3) If the Reactor Coolant System leakage exceeds the limits in either c.(1)(b) or c.(1)(c) above, the leakage rate must be reduced to within limits within 4 hours or the reactor must be placed in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. Leakage Into The Containment Free Volume

(1) Whenever the reactor is above cold shutdown, the total leakage into the containment free volume from both reactor coolant and non-reactor coolant sources combined shall not exceed 10 gpm.

(2) Notwithstanding the action which may be required by Specification 3.1.F.2.d.(3) below, with the combined leakage into the containment free volume greater than the above limit, the leakage rate must be reduced to within the specified limit within 12 hours or the reactor must be placed in cold shutdown within the following 36 hours.

(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; 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 within the recommended sensitivity guidelines of Regulatory Guide 1.45.

b. The containment radiogas monitor.

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 certain 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 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 by two separate and independent level switches, each of which actuates an audible alarm in the control room.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances. Determined leakage rates are an average over the applicable surveillance interval. Industry experience has shown that while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in cold shutdown. Primary system leakage through packing, gaskets, seal welds or mechanical joints is not considered to be PRESSURE BOUNDARY LEAKAGE.

The leakage limit and surveillance testing for RCS/RHR Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS/RHR Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by limitation of steam generator leakage between the reactor coolant system and the secondary coolant system. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown and the leaking tube(s) will be located and plugged.

The 10 gpm limit for combined reactor coolant and non-reactor coolant leakage into the containment free volume provides

allowance for a limited amount of leakage from sources other than the reactor coolant system within containment while conservatively limiting total leakage into the containment free volume to the same limit (i.e., 10 gpm) for identified reactor coolant leakage alone. This leakage is within the capabilities of the leakage detection and waste processing system and will not interfere with the detection of independent unidentified reactor coolant system leakage.

For those circumstances where high energy line failures occur inside containment resulting in flooding of the containment building sumps and/or floor, automatic actuation of reactor protection, safety injection and/or containment spray systems places the plant in a safe condition and, in some cases, provides intended flooding of the containment building. However, for those circumstances resulting from leakage or failure of low energy systems such as service water or component cooling inside containment, operator action is necessary to prevent accumulation of water on the containment floor to undesirable levels.

If the 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 is placed in cold shutdown within the next 36 hours. 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", the operator will immediately bring the reactor subcritical and initiate an expeditious cooldown of the plant.

The above actions are necessary to (1) preclude accumulation of water inside containment so that if a LOCA were to occur safety-related equipment would not become submerged, (2) prevent the reactor cavity from becoming filled with water, (3) prevent the reactor vessel from being wetted while it is at an elevated temperature, and (4) prevent the immersion of the in-core instrument conduits. The amount of water estimated to be inside containment after actuation of the emergency core cooling system following a loss of coolant accident is approximately 423,000 gallons. This amount of water would, by itself, reach approximately EL. 50' 1". An additional 28,000 gallons (a total of approximately 451,000 gallons) would have to accumulate inside containment before any safety-related electrical component would be submerged (approximately EL. 50' 5"). The combined volume of the containment sump, the recirculation sump and the containment

floor trenches is approximately 18,000 gallons. Since operator action is required by these specifications to shut the reactor down before these volumes are filled, sufficient margin between the water level inside containment following a loss of coolant accident and the level at which a safety-related electrical component may become submerged is maintained. Furthermore, since both sumps, the floor trenches and the containment floor up to EL. 46' 5 3/8" must be flooded (i.e., approximately 50,000 gallons) before the water level is sufficiently high to flood over the curb leading to the reactor cavity, the forementioned operator actions taken to preclude excessive flooding plus LOCA water levels will conservatively preclude flooding of the reactor cavity and subsequent wetting of the reactor vessel at an elevated temperature.

References

FSAR Sections 6.7, 11.2.3 and 14.2.4

Attachment B

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September 23, 1988

Safety Assessment

The specific proposed changes, as set forth in Attachment A to our Application, revises the LCO action and basis for the containment radioactivity monitors. The basis change (formerly .025 gpm to 10 gpm) now states that the leakage detection system will be capable of leakage detection within the recommended sensitivity guideline of Reg. Guide 1.45. This change is in accordance with Reg. Guide 1.45 and is more conservative than GL 84-04. The original TS requirements were based on optimum performance of the equipment and not on design necessities or regulatory requirements. The proposed LCO action statement will impose additional requirements which are more restrictive than at present.

Plant-specific analyses have been performed and are included in the Con Edison Leak Before Break submittal dated May 23, 1988. These analyses substantiate the use of the LBB Technology at IP-2 and provide a basis for this TS change. However, it should be noted that even without the use of these analyses, the TS as presently written are more restrictive than the Westinghouse Standard TS and than necessary.

Additionally, Attachment A contains changes to section 3.1.F of Indian Point Unit No. 2 Technical Specifications that correct typographical errors, make minor editorial changes and repaginate the text for spatial uniformity, as detailed in attachment C of this submittal.

Basis for No Significant Hazards Considerations Determination:

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration: exists by providing examples in 51 FR 7751. This amendment request falls under Examples II and VII of the Commission's Examples of Amendments That Are Considered Not Likely to Involve a Significant Hazards Consideration (51 FR 7751). In the case of the LCO action, it is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specification, e.g., a more stringent surveillance requirement (example II). In the case of the TS basis, it is a change to conform to changes in the regulations, where the license change results in very minor changes to facility operations that are clearly in keeping with the regulations (example VII).

In accordance with the requirements of 10 CFR 50.92, the proposed Technical Specification change is deemed to involve no significant hazards considerations because operation of Indian Point Unit No. 2 in accordance with this change would not:

1) Involve a Significant Increase in the Probability of Consequences of an Accident Previously Evaluated:

   The proposed changes to TS 3.1.F (Reactor Coolant System Leakage and Leakage into Containment Free Volume) merely make the Indian Point Unit 2 leakage detection system requirements for the containment radioactivity monitors consistent with those of Regulatory Guide 1.45.

This Reg. Guide has been endorsed by the staff on a generic basis, and it is now applied to IP-2. The analysis supporting the application of Reg. Guide 1.45 to IP-2 has been provided to the Staff in the LBB application submittal on May 23, 1988. It is apparent, therefore, that this change does not involve a significant increase in the probability or the consequences of an accident previously evaluated, but applies an accepted standard (Regulatory Guide 1.45) to substantiate the basis of this TS. This TS as written is unnecessarily restrictive and does not provide any additional safety benefits. The proposed change to the LCO action statement, 3.1.F.1.b.(6), imposes additional requirements on containment radioactivity monitors and does not increase the probability or consequence of an accident previously evaluated.

The additional changes requested are purely administrative in nature and only change typographical errors, make minor editorial changes for consistency, repaginate the document and delete pertinent portions of the IP-2 Technical Specification that are no longer effective or have been previously approved for deletion, therefore these changes do not increase the probability or consequences of an accident previously evaluated.

2) Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated:

The request to amend the TS merely allows conformance to Regulatory Guide 1.45 and imposes more restrictive requirements on the containment radioactivity monitors; it does not affect the reliability or the adequacy of the leakage detection system currently at IP-2, nor does it affect the design basis as described in the FSAR, in and of itself. The related submittal, which requests authorization for the use of the LBB methodology as specified in the final rule, also does not alter the existing plant design, but merely eliminates the necessity of considering the dynamic effects of Double Ended Guillotine Break (DEGB) on the primary system. Neither this TS changes nor the basis for these changes (LBB/Reg. Guide 1.45) creates the possibility of a new or different kind of accident, because the design itself is not being changed by this TS. The TS is merely being updated to include current accepted standards.

The additional changes requested are purely administrative in nature and only change typographical errors, make minor editorial changes for consistency, repaginate the document and delete pertinent portions of the IP-2 Technical Specification that are no longer effective or have been previously approved for deletion, therefore these changes do not create the possibility of a new or different kind of accident.

3) Involve a Significant Reduction in Margin of Safety:

Revising the LCO action and basis for the radioactivity monitors has a negligible effect on the margins of safety. The purpose of the containment air particulate monitors is to detect sufficiently small amounts of reactor coolant leakage to indicate an unacceptable plant condition, e.g., pipe cracks or excessive valve or seal leakage. The

accepted level of sensitivity sought by the staff is as set forth in Reg. Guide 1.45, and this TS change will allow IP-2 to conform to this accepted guidance. The IP-2 TS basis as set forth at present is based on optimum instrument function, not safety requirements, and thus is far too restrictive. Changing the TS to conform with Reg. Guide 1.45 will not involve a significant reduction in the margin of safety.

The additional changes requested are purely administrative in nature and only change typographical errors, make minor editorial changes for consistency, repaginate the document and delete pertinent portions of the IP-2 Technical Specification that are no longer effective or have been previously approved for deletion, therefore these changes do not involve a significant reduction in the margin of safety.

Since the application for amendment satisfies the criteria specified in 10 CFR 50.92 and is similar to examples for which no significant hazards consideration exist, Consolidated Edison Company has made a determination that the application involves no significant hazards considerations.

The proposed changes have been reviewed by the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant consideration.

Attachment C

Detailed List of Editorial Changes

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
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Present Page(s)	Present Section(s)	Change(s)
3.1.F-1	Top of Page	Changed "3.1.F." to "F" for consistency. Changed "Specification" to "Specifications".
3.1.F-1	3.1.F.1	Decapitalized title.
3.1.F-1	3.1.F.1.a	Decapitalized first letter of the first word of subsections (1) through (6)(c). Also, changed periods to commas as appropriate.
3.1.F-1	3.1.F.1.b	Capitalized the "s" in "specification".
3.1.F-1	3.1.F.1.b.(1)	Changed "...provided that on a daily basis the other..." to "...provided that, on a daily basis, the other..."
3.1.F-1	3.1.F.1.b.(3), (4) and (5)	Moved to page 3.1.F-2 as a result of line spacing adjustments.
3.1.F-2	3.1.F.1.b.(6)	Capitalized the "s" in "specification" in two places.
3.1.F-2	3.1.F.1.c	Capitalized the "s" in "specification". Deleted the comma after "then".
3.1.F-2	3.1.F.2	Decapitalized the title.
3.1.F-2	3.1.F.2.a	Changed "Leakage:" to "Leakage".
3.1.F-2	3.1.F.2.a.(1)	Changed "with 24 hours" to "within 24 hours".

Present Page(s)	Present Section(s)	Change(s)
3.1.F-2	3.1.F.2.a.(3)	Changed "shutdown" to "shut down".
3.1.F-2	3.1.F.2.b	Deleted the ":" in the title.
3.1.F-2	--	Bottom portion of page moved to page 3.1.F-3 as a result of line spacing adjustments.
3.1.F-2	3.1.F.2.b.(1)	Changed "&" to "and" and added a comma.
3.1.F-3	3.1.F.2.b.(2)	Capitalized the "s" in "specification".
3.1.F-3	3.1.F.2.c and d	Deleted the ":" in the titles and moved to page 3.1.F-4 as a result of line spacing adjustments.
3.1.F-3	3.1.F.2.d.(2)	Capitalized the "s" in "specification".
3.1.F-4	3.1.F.2.d.(4) and Basis	Moved to page 3.1.F-5 as a result ofline spacing adjustments.
3.1.F-4	3.1.F.2.d.(4)	Changed "39'-9"" to "39'9"".
3.1.F-4	1st Para. of Basis	Added a comma after "boundary or not".Changed "problem; and therefore, first indications" to "problem; therefore, first indications".
3.1.F-5	d.; 9th line	Changed "certian" to "certain".
3.1.F-5	d.; 18th line	Changed "Thus, monitoring of both the flow indication systems will..." to "Thus, monitoring of both flow indication systems will...".

Present Page(s)	Present Section(s)	Changes
3.1.F-5	e.; 15th line	Changed "...in the control room and two separate..." to "...in the control room and by two separate..." and added a comma after "switches".
3.1.F-5	Portion of c., d., and e.	Moved to page 3.1.F-6 as a result of line spacing adjustments.
3.1.F-6	First 5 Para.	Moved to page 3.1.F-7 as a result of line spacing adjustments.
3.1.F-6	4th Para., 2nd line	Added a comma after "integrity".
3.1.F-6	6th Para.	Moved to page 3.1.F-8 as a result of line spacing adjustments.
3.1.F-7	2nd Para.	Added commas after "El. 45'", and "El. 35'". Also changed "El. 39'-9"" to "El. 39'9"" and "El. 20'-5"" to "El. 20'5"".
3.1.F-7	3rd Para., 1st line	Deleted the ":".
3.1.F-7	3rd Para., 2nd line	Changed "such that" to "so that".
3.1.F-7	3rd Para., 11th line and 14th line	Changed "El. 50'-1" "to "El. 50'1"", and "El. 50'- 5"" to "El. 50'5"".
3.1.F-7	Portion of 3rd Para.	Moved to page 3.1.F-9 as a result of line spacing adjustments.
3.1.F-7	Last line	Changed "El. 46'-5 3/8"" to "El. 46' 5 3/8"".
3.1.F-8	1st line	Changed "prior to the water level being sufficiently high..." to "before the water level is sufficiently high...".