ATTACHMENT A

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March, 1987

3.4 STEAM and POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and City Water System operation is necessaary to ensure the capability to remove decay heat from the core.

Specification

A. The reactor shall not be heated above 350°F unless the following conditions are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Three auxiliary feedwater pumps each capable of pumping a minimum of 300 gpm must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank and a backup supply from the city water supply.

(4) Required system piping, valves, and instrumentation directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 uCi/cc.

B. Except as modified by 3.4.C below, if any of the conditions of 3.4.A above cannot be met within 72 hours*, the reactor shall be placed in the hot shutdown condition within the next 12 hours and subsequently cooled below 350°F using normal operating procedures.

ATTACHMENT B

Safety Assessment

Consolidated Edison Company of New York, Inc.
Indian Point No. 2
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Safety Assessment

The proposed technical specification revisions contained in Attachment A would reduce the minimum required pumping capability of the auxiliary feedwater pumps from 400gpm to 300gpm. The purpose of the change is to provide additional operational flexibility.

The safety function of the auxiliary feedwater pumps is to maintain a water inventory in the steam generators to remove core decay heat energy from the reactor coolant system in the event that the main feedwater system is inoperable. The FSAR Loss of Normal Feedwater transient provides the basis for determining the minimum auxiliary feedwater flow requirement. Therefore, this transient was reanalyzed with a constant 300gpm auxiliary feedwater flow rate and is included as Enclosure 1 to this Attachment. The analysis shows that with an auxiliary feedwater flow of 300gpm, sufficient feedwater is available to dissipate decay heat without water relief from the primary system relief or safety valves and that the primary system variables never approach a departure from nucleate boiling condition. Hence, the minimum auxiliary feedwater flow rate can be reduced from 400gpm to 300gpm without having any adverse effects upon the health and safety of the public.

Technical Specification 3.4.A.2 will be revised to change the minimum flow for each auxiliary feedwater pump from 400gpm to 300gpm.

Basis For No Significant Hazards Consideration Determination:

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870). Example (vi) of those involving no significiant hazards considerations discusses a change which may reduce a safety margin but where the results are clearly within all acceptable criteria with respect to the system or component. The proposed change to reduce the minimum auxiliary feedwater flow requirement is in a less restrictive direction and would appear to reduce a safety margin. However, consistent with the Commission's criteria in 10 CFR 50.92 (48FR71), we have determined that the proposed change does not involve a significant hazards consideration because the operation of Indian Point Unit No. 2 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change does not involve any physical change in plant equipment. The proposed revision is based on a conservative analysis which shows that with an auxiliary feedwater flow rate of 300gpm, sufficient feedwater would be available to dissipate decay heat without water relief from the primary system relief or safety valves, and that the primary system variables never approach a departure from nucleate boiling condition. Thus, the same safety criteria as previously evaluated are still met with the proposed change. Therefore, this change will not increase the probability or consequences of an accident.

(2) create the possibility of a new or different kind of accident from any accident previously evaluated, since the proposed change would not alter the configuration of any of the plant's equipment and the performance characteristics of the auxiliary feedwater system, conservatively construed, demonstrate that the system is fully capable of removing decay heat.

(3) involve a significant reduction in a margin of safety, since with the proposed change all safety criteria previously evaluated are still met and remain conservative.

Therefore, based on the above considerations, we conclude that the proposed change does not constitute a significant hazards consideration.

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 hazards consideration.

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Enclosure 1

To

Attachment B

Loss of Normal Feedwater Analysis

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
March, 1987

FSAR SECTION 14.1.9 (REVISED)

14.1.9 LOSS OF NORMAL FEEDWATER

14.1.9.1 Description

A loss of normal feedwater (from a pipe break, pump failure, or valve malfunction) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, primary plant damage could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not furnished, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur. A loss of significant water from the reactor coolant system could conceivably lead to core damage. Since the plant is tripped well before the steam-generator heat transfer capacity would be reduced, the primary system variables never approach a departure from nucleate boiling condition.

The following provides available protection against a loss of normal feedwater:

1. Reactor trip on low-low water level in any steam generator.

2. Reactor trip on steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator.

3. Two motor driven auxiliary feedwater pumps (400 gpm design capacity each) which are started on:

   a. Low-low level in any steam generator.
   b. Trip of any main feed pump turbine.
   c. Any safety injection signal.
   d. Manually.
   e. Loss of offsite power concurrent with unit trip.

4. One turbine-driven pump (800 gpm design capacity) which is started on:

   a. Low-low level in any two steam generators.
   b. Loss of offsite power concurrent with unit trip and no safety injection signal.
   c. Manually.

The motor-driven auxiliary pumps are powered by the diesels if a loss of off-site power occurs and the turbine-driven pump uses steam from the secondary system. Both types of pumps receive start signals within 1 min after the start of the accident. The turbine exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction from the condensate storage tank for delievery to the steam generators.

The above units provide considerable backup in equipment and control logic to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater, including that followed by loss of offsite power.

14.1.9.2 Method of Analysis

The analysis has been performed to show that following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat to prevent water relief through the pressurizer relief valves.

The following assumptions were made:

1. The initial steam-generator water level (in all steam generators) at the time reactor trip occurs is at the lowest level that will result in reactor trip and automatic initiation of auxiliary feedwater flow. Reactor trip and auxiliary feedwater system operation is assumed to be initiated by low-low steam generator level (0% of narrow range span). No credit is taken for the remaining actuation devices (e.g. steam flow-feedwater flow mismatch in coincidence with low steam generator level) for this analysis.

2. The plant is initially operating at 102 percent of 2758 MWt (the current Indian Point 2 power rating).

3. A conservative core residual heat generation based on the 1979 version of ANS-5.1 is used assuming long-term operation at the initial power level preceding the trip.

4. Only one motor-driven auxiliary feedwater pump is available at 1 min after the accident delivering 300 gpm.

5. A conservatively low heat transfer coefficient exists in the steam generator assuming reactor coolant system natural circulation.

6. Secondary system steam relief is through the self-actuated safety valves (Steam relief will, in fact, be through the steam line power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, these were conservatively not assumed available in the analysis.)

7. Coastdown of all reactor coolant pumps.

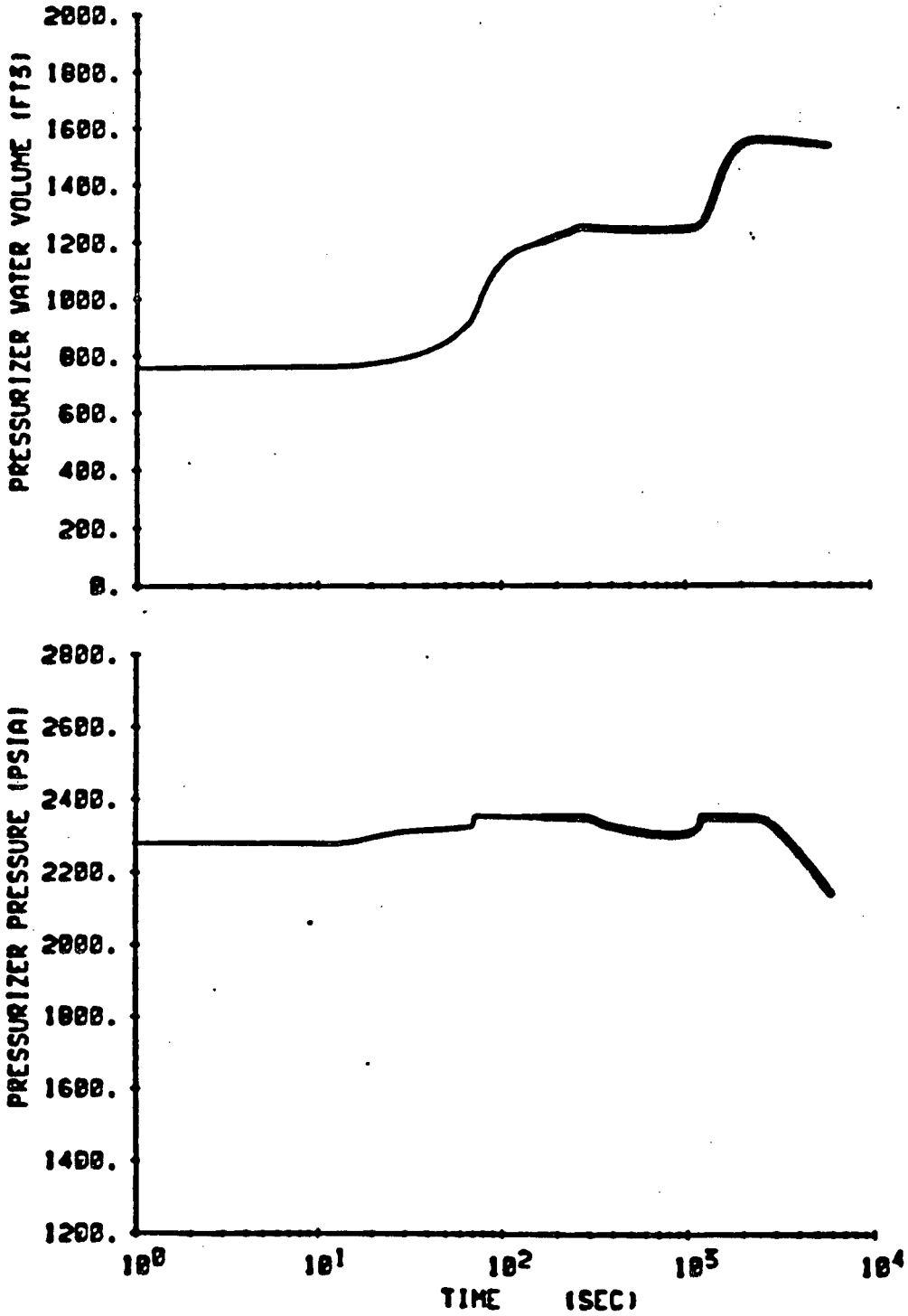
8. The pressurizer power-operated relief valves and pressurizer spray system are assumed to operate normally. This results in a conservative transient with respect to peak pressurizer water level. By assuming sprays, additional mass is injected into the pressurizer. If these control systems did not operate the pressurizer safety valves would maintain peak RCS pressure at or below the actuation setpoint (2500 psia) throughout the transient and the peak pressurizer water level would be lower. Thus, the pressurizer PORV's and spray system are not required to satisfy the safety criteria for this transient.

14.1.9.3 Results

Figures 14.1-51 A, B and C show the plant parameters following a loss-of normal feedwater accident with the assumptions listed above. Following the reactor and turbine trip from full load, the water level in the steam generator will fall because of the reduction of steam generator void fraction and because steam flow continues to dissipate the stored and generated heat. One minute following the beginning of the accident the auxiliary feedwater pump is automatically started, reducing the rate of water-level decrease. The capacity of the auxiliary feedwater pump is such that the water-level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valves.

From Figure 14.1-51A it can be seen that at no time is there water relief from the pressurizer. The assumption of more auxiliary feedwater capacity than that of one motor-driven pump or that the initial water level of one steam generator is above the low-low level trip level will of course result in increased margin to the point at which reactor coolant water relief could occur.

The results presented are for a reanalysis of the loss-of-normal-feedwater transient, which accounts for the current power rating (2758 MWt), and a constant auxiliary feedwater flow rate of 300 gpm.



CONSOLIDATED EDISON CO. INDIAN POINT UNIT 2
Figure 14.1-51 A Loss of Normal Feedwater with 300 gpm Auxiliary Feedwater

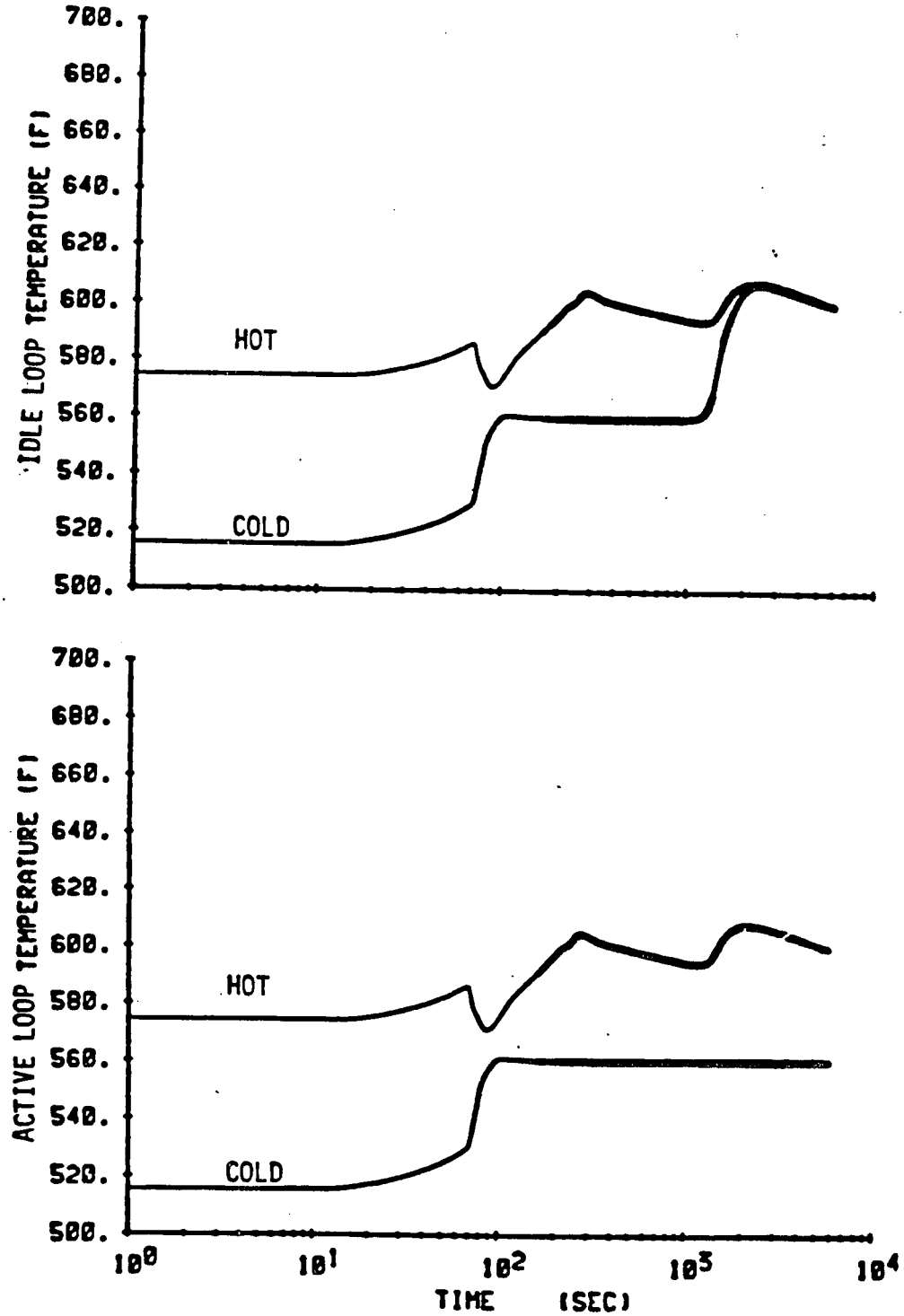


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Figure 14.1-51B

Loss of Normal Feedwater with 300 gpm Auxiliary Feedwater

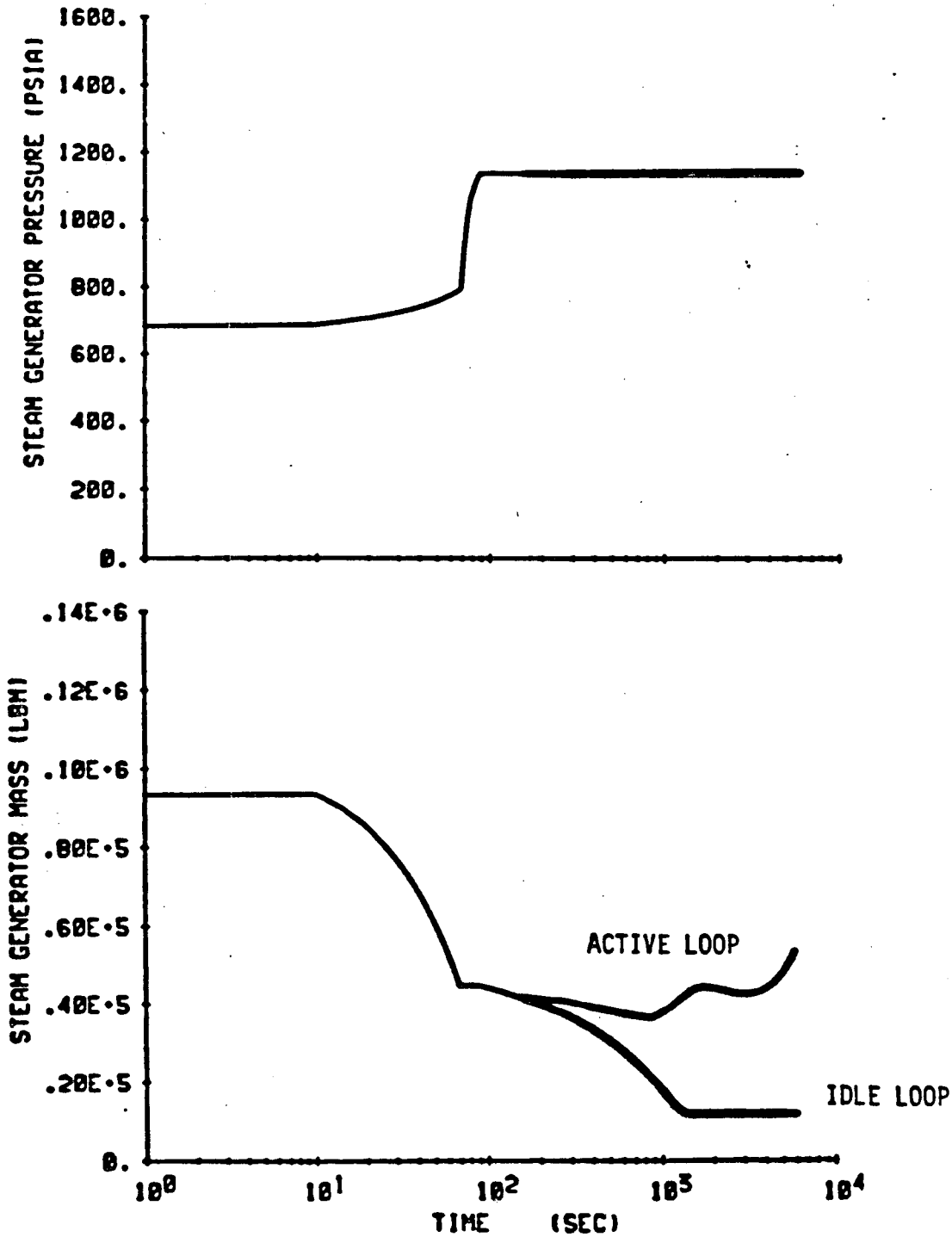

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Figure 14.1-51C

Loss of Normal Feedwater with 300 gpm Auxiliary Feedwater