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January 12, 1978

Docket No. 50-286

Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

We have reviewed your letters dated March 21 and June 15, 1977, which contained analyses of the potential consequences of a postulated fuel handling accident inside containment for Indian Point 3. These letters were in response to our letters of January 17, and May 5, 1977. In our letter of January 17, 1977, you were requested to review your technical specifications related to this postulated accident and, as appropriate, propose changes which would provide assurance that parameters important in the evaluation are maintained at levels which would assure that conservatively calculated offsite consequences were appropriately within the exposure guidelines of 10 CFR Part 100 over the facility lifetime. You were also requested to provide an evaluation of the consequences of this accident in which a single failure is assumed and propose any changes to facility equipment (e.g., redundant radiation monitors) which are necessary to assure that Part 100 guidelines are not exceeded. Our letter of May 5, 1977 was a request for additional information. Based on the information you have provided we conclude that additional information using more conservative assumptions is required to complete our review.

We conclude that there is not adequate assurance that the radiation monitoring system which automatically isolates containment upon high radiation will perform its function before a complete release of airborne radioactivity to the environs could take place. In addition, insufficient justification has been provided that 25000cf of mixing in containment building will occur prior to discharge of radioiodines. We also conclude that communication from the operators near the pool to the control room and subsequent remote manual actuation of isolation valves does not give adequate assurance of containing the radioactivity.

The radiation monitoring system intakes are in fan coolers which are not required to be operating during fuel handling operations in containment and the monitoring system does not meet single failure criteria. Further, the monitors may not be seismically qualified. The operator actions required may well take longer than the minimum time for the radioactivity to be exhausted from the containment.

The physical arrangement of the air ventilation equipment in the containment does not support the mixing inside containment assumed in your letter of june 15, 1977. The location of the fan coolers, the air intakes to these coolers, the recirculation ducts from these coolers and the purge exhaust duct do not assure that air flow patterns will in all possible cases mix activity from the damaged fuel assembly in some volume above the refueling cavity. This mixing is needed to keep the potential exposures from a postulated fuel handling accident inside containment (assuming the guidelines of Regulatory Guide 1.25) less than the 10 CFR Part 100 guidelines. A conservative but possible scenario for this accident is that a puff of radioactivity released from the damaged fuel assembly may travel directly from the refueling cavity to the ventilation exhaust duct and be discharged to the environs without any mixing within the containment. The location of the air intakes to the five recirculation fans and the purge exhaust duct appear to not preclude this possibility.

Possible means to provide adequate assurance that conservatively calculated offsite consequences are within the guidelines of 10 CFR Part 100 are: (1) increase the minimum time after shutdown before refueling, (2) redundant radiation monitors on the operating floor which will automatically isolate the containment, (3) a safety grade duct and charcoal filter on the purge exhaust from the containment, (4) smoke tests or other experiments or analysis which will demonstrate that the radioactivity released from the damaged fuel assembly would be mixed in the containment, or (5) conservative analysis which demonstrates that the containment would be isolated in a timely manner by the existing monitors assuming a single failure. The fifth approach should also include proposed technical specifications that will ensure that the recirculation fans and associated containment isolation monitors will be operating during fuel handling operations.

You are requested to provide analysis or propose facility modifications or technical specification changes which provide adequate assurance that the potential consequences are well within the guidelines of 10 CFR Part 100.

Response within 30 days is requested to allow us to complete our review prior to the first refueling of Indian Point Unit No. 3.

Sincerely,

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

cc: See next page

1/9/78 With changes included

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DATE	1/9/78	1/9/78	1/12/78			

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F122

January 10, 1978

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Dockets Nos.: 50-247
and 50-286

Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

RE: INDIAN POINT UNITS NOS. 2 AND 3

By letter dated November 17, 1976, we sent you a document entitled "NRC Staff Guidance for Complying with Certain Provisions of 10 CFR 50.55a(g), Inservice Inspection Requirements". In addition to clarifying the proper methods for complying with the regulation, this guidance provided a general outline of the type of information that the NRC staff would need to review inservice inspection and testing programs, and to evaluate requests for relief from ASME Code requirements that are determined to be impractical for a facility.

After reviewing a number of submittals relating to s50.55a(g) requirements from various licensees, we have concluded that additional guidance would be useful to all licensees to aid in the preparation of these submittals, and to expedite the NRC staff review and approval of the proposed programs and any requests for relief from certain ASME Code requirements. The need for this guidance is particularly evident for the pump and valve testing requirements.

Enclosed for your use is the "NRC Staff Guidance for Preparing Pump and Valve Testing Program Descriptions and Associated Relief Requests Pursuant to 10 CFR 50.55a(g)". This enclosure defines the required scope of a pump and valve testing program, itemizes the specific information needed for staff review, and provides guidelines for submitting information to support requests for relief from any ASME Code requirements found to be impractical for a facility. The same information is being sent to all nuclear power plant licensees and is intended to complement and expand on the guidance we provided to

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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you in our previous letter. Although the enclosure specifically addresses pump and valve testing requirements only, the same level of detailed information identified in this guidance should also be provided in inservice inspection program submittals.

We request that you follow the enclosed guidance to the greatest extent possible when submitting proposed inservice inspection and testing programs and requests for relief from ASME Code requirements, and when responding to additional information requests from the staff. Your adherence to this guidance will minimize the NRC staff review time needed to approve your proposed programs and associated relief requests.

If you have any questions regarding implementation of 10 CFR 50.55a(g) at your facility, please contact us.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosure:
NRC Staff Guidance

cc w/enclosure:
See next page

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NRC STAFF GUIDANCE FOR PREPARING PUMP AND VALVE TESTING PROGRAM DESCRIPTIONS AND ASSOCIATED RELIEF REQUESTS PURSUANT TO 10 CFR 50.55a(g)

The guidance provided in this enclosure is intended to illustrate the type and extent of information that should be provided in proposed pump and valve testing program descriptions and to support associated requests for relief from ASME Code requirements. By utilizing these guidelines, licensees can significantly reduce the need for having to respond to additional information requests from the NRC staff.

I. Pump and Valve Testing Program Description

A. Scope of the Program:

1. The pump testing program should include all safety related* Class 1, 2 and 3 pumps that are provided with an emergency power source.

2. The valve testing program should be limited to the safety related* valves. All such valves must be addressed in the program and should include, as a minimum, those in the following systems. Valves in these systems which are used for operating convenience only - such as manual vent, drain, instrument and test valves, and valves used for maintenance only should be excluded.

   For PWR's:

   a. High Pressure Injection System

   b. Low Pressure Injection System

   c. Accumulator Systems

   d. Containment Spray System

   e. Primary and Secondary System Safety and Relief Valves

   f. Auxiliary Feedwater Systems

*Safety related are those pumps and valves necessary to safely shut down the plant or mitigate the consequences of an accident.

g. Reactor Building Cooling System

h. Active Components in Service Water and Instrument Air Systems which are required to support safety system functions

i. Containment Isolation Valves that are required to change position on a containment isolation signal

j. Chemical and Volume Control System

k. Other key valves in Auxiliary Systems which are required to operate to directly support plant shutdown or safety system function; such as, emergency diesel starting air valves, component cooling water supplies, etc.

l. Residual Heat Removal System

m. Reactor Coolant System

For BWR's:

a. High Pressure Coolant Injection System

b. Low Pressure Coolant Injection System

c. Residual Heat Removal System (Shutdown Cooling System)

d. Emergency Condenser System (Isolation Condenser System)

e. Low Pressure Core Spray System

f. Containment Spray System

g. Safety, Relief, and Safety/Relief Valves

h. RCIC (Reactor Core Isolation Cooling) System

i. Containment Cooling System

j. Containment isolation valves that are required to change position on a containment isolation signal

k. Standby liquid control system (Boron System)

l. Automatic Depressurization System (any pilot or control valves, associated hydraulic or pneumatic systems, etc.)

m. Control Rod Drive Hydraulic System ("Scram" function)

n. Other key valves in Auxiliary Systems which are required to operate to directly support plant shutdown or safety system function; such as, emergency diesel starting air valves, component cooling water supplies, etc.

o. Reactor Coolant System

B. The following information should be provided for NRC staff review of the Pump and Valve Testing Programs:

1. Three sets of P&ID's, that are large and clear enough to be read easily, and which include all of the systems listed above, with the ASME code class and system boundaries clearly marked. The drawings should include all of the components present at the time of submittal and a legend of the P&ID symbols.

2. Identification of the applicable ASME Code Section XI Edition and Addenda.

3. The period for which the program is applicable.

4. Identification of the component ASME Section III Code Class.

5. For Pump testing, identification of:

   a. Each pump required to be tested (name and number)

   b. The test parameters to be measured

   c. The test frequency

6. For valve testing, identification of:

   a. Each valve in ASME Section XI Categories A and B that will be exercised every three months during normal plant operation (indicate whether partial or full stroke exercise, and for power operated valves list the limiting value for stroke time).

   b. Each valve in ASME Section XI Category A that will be leak tested during refueling outages (indicate the leak test procedure you intend to use).

   c. Each valve in ASME Section XI Categories C and D that will be tested, the type of test and the test frequency. For check valves, identify those that will be exercised every 3 months and those that will only be exercised during cold shutdown or refueling outages.

   d. Each valve in ASME Section XI Category E that will be operationally checked.

   e. The following additional information, if practical:

      i. The valve location coordinates or other appropriate location information which will expedite locating the valves on the P&IDs.

      ii. Identification of all valves that are provided with an interlock to other components and a brief description of that function.

II. Requests for Relief from Certain Pump or Valve Testing Requirements

It has been the staff's experience that many requests for relief from testing requirements, submitted by licensees, have not been supported by adequate descriptive and detailed technical information. This detailed information is necessary to document why the burden imposed on the licensee in complying with the code requirements is not justified by the increased level of safety obtained from the testing.

Relief requests which are submitted with a justification such as "impractical", "inaccessible", or any other categorical basis, require additional information to allow the staff to make an evaluation of that relief request. The intention of the guidance

set forth below is to illustrate the extent of the information that is required by the NRC staff to make a proper evaluation and to adequately document the basis for granting the relief in the safety evaluation report. The NRC staff believes that if this information is provided in the licensee's submittal, subsequent requests for additional information and delays in completing the review, and granting the relief, can be considerably reduced.

A. Specific information required for NRC review of requests for relief from testing requirements:

1. Identification of the component for which relief is requested:

   a. Name and number as given in FSAR

   b. Function

   c. ASME Section III Code Class

   d. For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000

2. Specific identification of the ASME Code requirement that has been determined to be impractical for each component.

3. Information to support the determination that the requirement in (2) is impractical; i.e., state and explain the basis for requesting relief.

4. Specification of the inservice testing that will be performed in lieu of the ASME Code Section XI requirements, if any.

5. The schedule for implementation of the procedure(s) in (4).

B. Examples to illustrate several possible areas where relief may be granted and the type and extent of information necessary to support the granting of relief:

1. "Accessibility":

   The regulation allows relief to be granted from code requirements because of insufficient access provisions. However, a detailed discussion of actual physical arrangement of the component in question to illustrate the insufficiency of space for conducting the required test is necessary.

In addition, discussion of the alternative surveillance techniques that have been considered should be provided. If these alternative techniques have been determined to be impractical, the basis for this determination should be provided.

2. "Environmental Conditions Prohibitive" (e.g., high radiation level, high temperature, high humidity, etc.):

   Although it is prudent to maintain occupation radiation exposure for inspection personnel as low as practicable, the request for relief from code requirements cannot be granted solely on the basis of high radiation levels.

   A balanced judgment between the hardships and compensating increase in the level of safety must be explicitly justified. Therefore, detailed information regarding the radiation levels at the required test location, along with estimated yearly man-rem exposures associated with the testing, should be provided. Alternative testing techniques that have been considered should be discussed. If these alternative techniques have been determined to be impractical, the basis for this determination should be provided.

3. "Instrumentation Not Originally Provided":

   Information to justify that installation of the needed instrumentation to comply with the code requirements would result in undue burden or hardships without a compensating increase in the level of plant safety should be provided. Alternative testing techniques that have been considered should be discussed. If these alternative techniques have been determined to be impractical, the basis for this determination should be provided.

4. "Valve Cycling During Plant Operation Could Put the Plant in an Unsafe Condition":

   A detailed explanation as to why exercising tests during plant operation could jeopardize the plant safety. Examples of the type of valve that the staff considers to be in this category are: valves whose failure in a non-conservative position during the cycling test would cause a loss of total system function; valves whose failure to close during the

cycling test would cause a loss of containment integrity; and valves, which when cycled, could subject a system to pressures in excess of their design pressures. A plant specific explanation must be provided.

5. "Valve Testing at Cold Shutdown or Refueling Intervals in lieu of the 3 Month Required Interval":

   The licensee should explain in detail why each valve cannot be exercised during normal operation. Also, for the valves where a refueling interval is indicated, the licensee should explain in detail why each valve cannot be exercised during each cold shutdown.

C. The following acceptance criteria for granting relief are utilized by the staff:

   The licensee must successfully demonstrate with documented information that:

   1. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety, and noncompliance will provide an acceptable level of quality and safety, or

   2. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

III. Standard Format for Valve Testing Submittals

A recommended standard format, for the valve portion of the pump and valve testing program and relief requests, is included as an attachment to this Guidance. The NRC staff believes that the use of this standard format would reduce the time spent by both the staff in its review, and by the licensee in their preparation, of the pump and valve testing program submittals. The standard format includes examples of relief requests which are intended to illustrate the application of the standard format only and are not necessarily applicable to any specific plant.

ATTACHMENT

RECOMMENDED STANDARD FORMAT FOR VALVE INSERVICE TESTING PROGRAM SUBMITTALS

SYSTEM NAME AUXILIARY COOLANT SYSTEM COMPONENT COOLING P&ID NO. 1045-E-2A PAGE _____

Valve Number	Class	Coordinates	Valve Category					Size (inches)	Valve Type	Actuator Type	Normal Position	Test Requirements	Relief Requests	Testing Alternative	REMARKS (Not to be used for relief basis)
			A	B	C	D	E								
710	3	D-14					X	4	GA	M	LO	ET			
700	3	D-15				X		6	DE	NA	C	DT			
717	3	C-15			X			16	CK	SA	-	CV	X	CS	
702C	3	C-15			X			16	CK	SA	-	CV			
707	3	E-14			X			3	REL	SA	-	CV			
834	3	D-11		X			X	4	GL	M	C	Q	X	ET	
												MT			60 sec. stroke time
722B	3	B-11			X			3/4	REL	SA	-	SRV			
722C	3	B-11			X			3/4	REL	SA	-	SRV			
715	2	A-10			X			3	REL	SA	-	SRV			
729	2	B-10			X			3	REL	SA	-	SRV			
744B	2	D-14	X					10	GA	MO	C	Q			
												LT	X		
												MT			30 sec. stroke time

LEGEND FOR VALVE TESTING EXAMPLE FORMAT

Q - Exercise valve (full stroke) for operability every (3) months

LT - Valves are leak tested per Section XI Article IWV-3420

MT - Stroke time measurements are taken and compared to the stroke time limiting value per Section XI Article IWV 3410

CV - Exercise check valves to the position required to fulfill their function every (3) months

SRV - Safety and relief valves are tested per Section XI Article IWV-3510

DT - Test category D valves per Section XI Article IWV-3600

ET - Verify and record valve position before operations are performed and after operations are completed, and verify that valve is locked or sealed.

CS - Exercise valve for operability every cold shutdown

RR - Exercise valve for operability every reactor refueling

RELIEF REQUEST BASIS

System: Auxiliary Coolant System, Component Cooling

1.

Valve:	717
Category:	C
Class:	3
Function:	Prevent backflow from the reactor coolant pump cooling coils
Test Requirement:	Exercise valve for operability every three months
Basis for relief:	To test this valve would require interruption of cooling water to the reactor coolant pumps motor cooling coils. This action could result in damage to the reactor coolant pumps and thus place the plant in an unsafe mode of operation.
Alternate Testing:	This valve will be exercised for operability during cold shutdowns

2.

Valve:	834
Category:	B-E
Class:	3
Function:	Isolate the primary water from the component cooling surge tank during plant operation. It is normally in the closed position, but routine operation of this valve will occur during refueling and cold shutdowns.
Test Requirement:	Exercise valve (full stroke) for operability every three (3) months.
Basis for Relief:	This valve is not required to change position during plant operation to accomplish its safety function. Exercising this valve will increase the possibility of surge tank line contamination.
Alternate Testing:	Verify and record valve position before and after each valve operation.

3.

Valve:	744B
Category:	A
Class:	2
Function:	Isolate the residual heat exchangers from the cold leg R.C.S. backflow and accumulator backflow.
Test Requirements:	Seat leakage test
Bases for relief:	This valve is located in a high radiation field of mr/hr which would make the required seat leakage test hazardous to test personnel. The estimated yearly man-rem exposure associated with performing the required seat leakage test is _____. We intend to seat leak test two other valves (875B and 866B) which are in series with this valve and which also prevent backflow. We feel that by complying the seat leakage requirements for 744B we will not achieve a compensatory increase in the level of safety.
Alternate Testing:	No alternative seat leak testing is proposed for 744B.