ATTACHMENT A

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Power Authority of the State of New York

Indian Point Unit No. 3
Docket No. 50-286
Facility Operating License No. DPR-64

December, 1977

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specification

A. Containment Integrity

1. The containment integrity (as defined in 1.10) shall not be violated unless the reactor is in the cold shutdown condition. However, those non-automatic valves listed in Table 3.6-1, may be opened if necessary for plant operation and only as long as necessary to perform the intended function. Non-automatic containment isolation valves may be added to plant systems without prior license amendment to Table 3.6-1 provided that a revision to this Table is included in a subsequent license amendment application.

2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin $\geq 10\% \frac{\Delta k}{k}$.

3. If containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within four hours or the reactor shall be brought to a cold shutdown condition within the next 36 hours, utilizing normal operating procedures.

B. Internal Pressure

If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

TABLE 3.6-1

NON-AUTOMATIC CONTAINMENT ISOLATION VALVES
OPEN CONTINUOUSLY OR INTERMITTENTLY FOR PLANT OPERATION

550	752F	SWN-41	SWN-44
744	753F	SWN-43	SWN-51
1870	752J		
743	753J	SWN-41	
732		SWN-43	SWN-71
885A			SWN-71
885B		SWN-41	SWN-71
205		SWN-43	SWN-71
226	863		SWN-71
227	878A 878B	SWN-41	
250A	PCV-1111	SWN-43	UH-37
241A	PCV-1111		UH-38
250B	1814A	SWN-41	1882A
241B	1814B	SWN-43	1875A
250C	1814C		1875B
241C	859A	SWN-44	1876A
250D	859C	SWN-51	1876B
241D	1833A	SWN-44	PS-7
869A	1833B	SWN-51	PS-8
869B	SA-24	SWN-44	PS-9
851A	SA-24	SWN-51	PS-10
850A	580A	SWN-44	888A
1610	580B	SWN-51	888B
990A	958		1890A
990B	959		1890B
	990C		1890C
			1890D
			1890E
			1890F
			1890G
			1890H
			1890J

3. Containment isolation valves may be added to plant systems without prior license amendment to Table 4.4-1 provided that a revision to this Table is included in a subsequent license amendment application.

F. Containment Modifications

Any major modification or replacement of components of the containment performed after the initial pre-operational leakage rate test shall be followed by either an integrated leakage rate test, or a local leak detection test and shall meet the appropriate acceptance criteria of A.2, C.2, or E.2. Modifications or replacements performed directly prior to the conduct of an integrated leakage rate test shall not require a separate test.

G. Report of Test Results

Each integrated leakage rate test shall be the subject of a summary technical report to be submitted to the Nuclear Regulatory Commission in accordance with the requirements of Appendix J to 10 CFR 50, effective issue date March 16, 1973. Each report shall include leakage test results and a summary analyses of sensitive leak rate, air lock, and containment isolation valve tests performed since the previous integrated leakage rate test.

H. Annual Inspection

A detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the containment structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number (1)	Test Fluid (2)	Minimum Test Pressure (PSIG)
241C	10	Water (4)	45
250D	10	Water (4)	45
241D	10	Water (4)	45
222	11	Water (4)	45
956E	12	Water (4)	45
956F	12	Water (4)	45
869A	14	Water (4)	45
867A	14	Gas	41
878A	14	Gas	41
869B	14	Water (4)	45
867B	14	Gas	41
878B	14	Gas	41
1835A	15	Nitrogen (4)	41
1835B	15	Nitrogen (4)	41
1833A	15	Water (4)	45
1833B	15	Water (4)	45
851A	15	Water (4)	45
850A	15	Water (4)	45
859A	16	Water (4)	45
859C	16	Water (4)	45
8406	17	Gas	41
863	17	Gas	41
956G	18	Water (4)	45
956H	18	Water (4)	45
1786	19	Water (4)	45
1787	19	Water (4)	45

CONTAINMENT ISOLATION VALVES

Valve No.	Penetration Number(1)	Test Fluid(2)	Minimum Test Pressure (PSIG)
1890A	57	Gas	41
1890C	57	Gas	41
1890F	57	Gas	41
1890B	57	Gas	41
1890G	57	Gas	41
1890H	57	Gas	41
1890J	57	Gas	41
1882A	58	Gas	41
IV-2A	58	Gas	41
IV-2B	58	Gas	41
1875A	59	Gas	41
IV-3A	59	Gas	41
1876A	60	Gas	41
IV-5A	60	Gas	41
1875B	61	Gas	41
IV-3B	61	Gas	41
1876B	62	Gas	41
IV-5B	62	Gas	41
IA-39	64	Gas	41
PCV-1228	64	Gas	41
PS-7	65	Gas(7)	41
PS-10	65	Gas(7)	41
PS-8	65	Gas(7)	41
PS-9	65	Gas(7)	41
CB-1 (EL.80')	69	Gas	41
CB-2 "	69	Gas	41
CB-3 "	69	Gas(7)	41
CB-4 "	69	Gas(7)	41
CB-1(EL.95')	69	Gas	41
CB-2 "	69	Gas	41
CB-3 "	69	Gas(7)	41
CB-4 "	69	Gas(7)	41

ATTACHMENT B

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Safety Evaluation

Consolidated Edison Company of New York, Inc.
Power Authority of the State of New York

Indian Point Unit No. 3
Docket No. 50-286
Facility Operating License No. DPR-64

December, 1977

SAFETY EVALUATION

The proposed changes to the Indian Point 3 Technical Specifications, contained in Attachment A to this Application, would reflect the addition of hardware installed inside containment for the Overpressure Protection System during the turbine maintenance outage. As part of this modification, check valve 8406 was installed in series with existing valve 863 to provide containment isolation for the new Overpressure Protection System. Check valve 8406 has been leak tested and qualified in accordance with Appendix J requirements. This check valve will also satisfy containment isolation requirements presently satisfied by valves 891A, B, C and D for the N_2 supply to the ECCS accumulators. The proposed changes would delete valves 891A, B, C and D from Tables 3.6-1 and 4.4-1 and add check valve 8406 to Table 4.4-1.

Additionally, Table 4.4-1 would be further modified to effect technical clarifications concerning leak rate testing of air lock containment isolation valves. The containment isolation valves in both the EL-80' and the EL-95' air locks were leak tested and qualified in accordance with Appendix J requirements. The proposed change would list the containment isolation valves for each air lock separately in Table 4.4-1.

To clarify the use of Tables 3.6-1 and 4.4-1, it has been explicitly stated in the specification for each table that containment isolation valves may be added without prior License

Amendment. A revision to the table would be included in a subsequent License Amendment Application. Valves can not be deleted from these tables without prior License Amendment.

The proposed changes do not in any way alter the safety analyses performed for Indian Point 3. The proposed changes have been reviewed by the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. Both committees concur that these changes do not represent a significant hazards consideration and will not cause any change in the types or increase in the amounts of effluents or any change in the authorized power level of the facility.

Docket 50-286

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

F118

August 4, 1978

All Power Reactor Licensees

Gentlemen:

This letter and enclosed NUREG-0219 titled "Nuclear Security Personnel for Power Plants, Content and Review Procedures for a Security Training and Qualification Program," dated July 1978, are being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with applications for a license to operate or construct a power reactor.

Within the next few weeks the Commission is scheduled to publish in final form amendments to 10 CFR 73 to impose upgraded qualification, training, and equipping requirements for security personnel protecting against theft of special nuclear material and industrial sabotage of nuclear facilities or nuclear shipments. The enclosed document provides a basis on which commercial nuclear reactor applicants and licensees can develop acceptable programs to implement these new requirements.

A second draft of this document was published for comment on April 21, 1978 and as a result the staff has considered the comments received and incorporated many changes. The following summarizes the major comments received and how the NRR staff addressed them in preparing the final document:

1. Approximately one third of the 32 that commented stated that the sample plan indicated an excessive amount of detail and the guidance should not exceed that currently given for safety related training.

   The final document contains only 25 pages of guidance (Parts 1&2); the remainder is a sample plan. The sample was provided to assist the applicants and licensees in preparation of a plan based on a new approach. As noted in item 3 below, the sample should not be considered a requirement.

   The staff reformated the sample plan to reduce the amount of detail and removed many tasks based on the ratings submitted in response to the request in Draft 2. This resulted in a reduction of 46% in the number of pages devoted to performance objectives (173 vs. 94) and a reduction of 44% in the number of performance objectives (344 vs. 191). A further reduction should be realized when the site analysis is completed, since the sample plan includes many tasks that are not appropriate for all sites.

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2. Many comments stated that the number of onsite evaluations by the NRC was excessive (i.e., 1 by NRR every 2 years and 3 each year by I&E).

   The I&E schedule set forth in the draft was based on the established frequency of onsite I&E physical security inspections with the assumption that these inspections would be expanded to include training and personnel qualification. However, all references to I&E inspection have now been deleted from the final version since this document addresses NRR policy only.

3. Some commented that although we state that each site is required to develop a qualification program based on a site specific job analysis, that the NRR reviewers would treat the sample plan in NUREG-0219 as the only acceptable approach.

   The NRR staff feels that the sample plan provides valuable guidance and should remain in the document. However, the final version was revised to stress that the sample is <u>not</u> a requirement. One example is found on page 1-1 and reads:

   > "It must be stressed that it is the responsibility of each site, using the methodology described in this document, to identify its site-specific tasks, elements, and performance objectives. The security program selected must evaluate each individual's ability to implement the site-approved physical security and contingency plans. Training and evaluation are not done for their own sake.
   >
   > The sample qualification plan found in part 3 should not be considered a requirement, but only a guide; Each specific site plan is reviewed on its own merits."

4. Other comments stated that tasks shown in the sample were too extensive. They indicated that the sample program exceeded that required by most military and police organizations and/or the requirements to meet the 73.55 threat level. A few commented that the type of response indicated in the sample plan is outside the responsibility and capabilities of private security.

   The applicants and licensees are required to identify in their qualification plan only those security tasks critical to successful implementation of the site contingency and physical security plans. If a licensee can develop acceptable contingency plans that meet the threat and do not require police or military tactics, then the tactical tasks can be deleted. However, it must be realized that the military and police are the only organizations with experience dealing with such problems. The vast majority of the military and police related tasks contained in the sample are at the basic training level.

5. Finally, a few commented that the NRC should hold working sessions with the utilities to develop its detailed requirements.

   Although the actual development of training and qualification plans are the responsibility of each licensee, NRR is planning to hold a series of workshops with the utilities to develop a mutual understanding of how to implement the methodology described in NUREG-0219. These workshops will be small and devoted to actual plan development.

Additional copies of NUREG-0219 can be obtained from the National Technical Information Service, Springfield, Virginia 22161 at current prices.

Sincerely,

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosure:
NUREG-0219

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Docket
50-286

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

F118

July 18, 1978

All Power Reactor Licensees and Applicants with Docketed Applications To Construct or Operate a Power Reactor

Gentlemen:

This letter and enclosed NUREG/CR-0181 are being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with applications for a license to operate or construct a power reactor.

NUREG/CR-0181 provides basic barrier and penetration data needed for physical security system assessment. The data provided in the NUREG is being used by NRC during the reactor safeguards review process.

Additional copies of this document can be obtained from the National Technical Information Service, Springfield, Virginia 22161, at current prices.

Sincerely,

James R. Miller

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosure:
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