POWER AUTHORITY OF THE STATE OF NEW YORK

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February 11, 1980
IPN-80-16

Mr. William Gammill, Acting Assistant Director
for Operating Reactor Projects
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Adequacy of Station Electric
Distribution Systems Voltages

Dear Mr. Gammill:

The purpose of this letter is to respond to Mr. Schwencer's letter dated July 27, 1979 and also to provide a partial response to your August 8, 1979 letter.

As discussed in your letter of August 8, 1979, the NRC has expanded its generic review of the adequacy of the electric power systems in addition to the concerns expressed in the June 3, 1977 correspondence with regard to degraded voltage conditions due to conditions originating on the grid. Your letter is effectively an extension of previous questions that have been asked and expands the scope of analyses that must be performed. This will require extensive computer analyses which are not anticipated to be completed until June 1, 1980.

Our response to Mr. Schwencer's questions on degraded grid voltage (Attachment 1) should serve the additional purpose of providing preliminary answers to your letter. The response to question 1 discusses these preliminary conclusions and the need for computer analyses to verify them.

With regard to the technical specifications requested by Mr. Schwencer's third question, we believe that it would be premature to propose any change to the technical specifications until we have reported the results of our computer studies in June 1980 and the staff has had time to review them.

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Very truly yours,

George M. Wilverding
for
Paul J. Early
Assistant Chief Engineer-Projects

cc: see attached sheet

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cc: Mr. T. Rebelowski
Resident Inspector
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ATTACHMENT 1

# RESPONSE TO NRC REQUEST
# FOR ADDITIONAL INFORMATION
# DEGRADED GRID VOLTAGE

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
FEBRUARY 5, 1980

QUESTION:

1. The voltage monitors located at non-Class 1E 6.9 KV buses and those located on the safeguard 480V buses are not designed to satisfy the requirements of IEEE Std. 279-1971. Further the voltage protection system does not include coincident logic to preclude spurious trips of the offsite power source.

RESPONSE:

Voltage monitors 27-2, 27-3, 27-5 and 27-6 monitoring 6.9kV buses 2,3,5 and 6 respectively are designed to meet the requirements of the single failure criteria on a bus to bus basis.

A failure of one of these relays (set at 81% voltage) will either:

a) (Failure to sense undervoltage) leave its single associated 480 volt bus connected to the preferred power supply until voltage decays to the setting of the 480 volt bus undervoltage relays (46%) or until the operator takes action to manually disconnect the preferred power supply by tripping the associated 6.9kV or 480 volt station service transformer breaker or

b) (Spurious operation) cause its single associated 480 volt bus to connect to its standby power supply before it is necessary.

Spurious operation of any of these relays would not result in the loss of capability for any protective action. It would only cause the single affected 480 volt bus to revert to a second contingency operable condition powered from the standby power supply.

Voltage monitors 27-1/2A, 27-2/2A, 27-1/3A, 27-2/3A, 27-1/5A, 27-2/5A, 27-1/6A and 27-2/6A similarly meet the requirements of the single failure criteria on a bus to bus basis.

We do recognize that the trip singals from the 6.9kV undervoltage relays accomplish their function by actuating trips of non-safety related 6.9kV switchgear breakers. We feel that the benefits provided by segregating the function of automatic isolation from the function of diesel sequencing in this manner outweigh any advantage that could be gained by reconnecting these trip signals to the safety related 480 volt switchgear breakers. The basis for our judgement is as follows:

1. In the existing design there is no potential for "marginal" undervoltage settings to interrupt diesel

sequencing due to inrush related voltage dips.

2. In the existing design there is no potential for control circuit interactions between the non-safety related 6.9kV switchgear and the safety related 480 volt switchgear.

3. Analyses performed by Con Edison have shown that the worst case single or multiple equipment contingency conditions for the Indian Point offsite power supply would not result in safety related bus voltages below these required for equipment operation (reference August 29, 1977 letter William J. Cahill, Jr. to Robert W. Reid response to question 1.F). Minimum voltages would be as reported in the referenced letter or the equipment contingency would result in a complete loss of voltage to the offsite buses.

4. The split bus configuration of the preferred power supply during normal operation (480 volt buses 5A and 6A fed from offsite power and 480 volt buses 2A and 3A "unitized" from the main generator output) provides effective independence of at least one power train from any abnormal conditions associated with the 138kV offsite supply, the main generator output or the station or unit auxiliary transformers.

5. Unacceptably low 480 volt bus voltage conditions resulting from any unusual system wide contingency on the Con Edison system for IP unit 3 would be immediately annunicated to the operator by new undervoltage relays set at 93.3% voltage on each of the four (4) safety related 480 volt buses. (Reference August 29, 1977 letter William J. Cahill, Jr. to Robert W. Reid, response to question 4). The operator could then take manual action to isolate the safety related buses from the preferred power supply.

As stated above, we feel that the existing design is more than acceptable. We are, however, in the process of performing additional studies to reconfirm the adequacy of our onsite distribution system based on considerations of the Commission's letter of August 8, 1979 regarding the event at Arkansas Nuclear One and the guidelines for voltage drop calculations (enclosure 2) contained therein.

While our preliminary review of this new material from a qualitative standpoint supports our previous conclusions with regard to the adequacy of the onsite distribution system, quantitative analyses are necessary to confirm this judgement. Some additional data accumulation (i.e. for the 13.8kV backup perferred power supply) calculations and more extensive documentation as requested by your guidelines will be required. In addition, we will have to re-generate the data base which was used for the original voltage study calculations since this was removed from computer memory during the two years since results of the original load flows were submitted to the Commission.

We estimate that this effort will be completed by June of 1980. This effort will be pursued with a similar high priority to the Three Mile Island related studies and projects which are ongoing at the Authority.

QUESTION:

2. The voltage monitors (with a setpoint of 81% at the 6.9 KV bus) do not project the safety related equipment for a sustained degradation of voltages at all onsite distribution levels. The 6.9 KV bus undervoltage relay setpoint reflected down to the 480V buses show voltages to be between 337V (70.3%) and 368V (76.7%). As stated in your letter dated August 29, 1977, these voltages are below the rating of the safety related electrical equipment.

RESPONSE:

Raising the 6.9kV undervoltage settings to levels necessary for these relays, by themselves, to protect the safety related buses, would prevent proper coordination of the preferred and standby power supplies. These high settings could result in transfer of <u>all</u> safety related 480 volt buses to their backup power supplies before it is desirable.

As outlined in the preceeding response, we feel that the existing design is more than acceptable, but further studies are in progress.

QUESTION:

3. Technical Specification changes to include limiting conditions of operation, surveillance requirements, trip setpoints with maximum and minimum limits and allowable values for the second level of voltage protection monitors have not been included in your previous submittals.

RESPONSE:

We believe that it would be premature to propose any change to the technical specifications until we have reported the results of our computer studies in June of 1980 and the staff has had time to review them.

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

February 11, 1980

Docket Nos. 50-3
50-247
and 50-286

Ms. Ellyn R. Weiss, Esquire
Sheldon, Harmon, and Weiss
1725 I Street, N.W., Suite 506
Washington, D. C. 20006

Dear Ms. Weiss:

By petition dated September 17, 1979, you requested on behalf of the Union of Concerned Scientists that the Nuclear Regulatory Commission revoke the Indian Point Unit 1 license, decommission Indian Point Unit 1, and suspend operations at Indian Point Units 2 and 3. As discussed in the enclosed Decision, the staff agrees that the provisional operating license for Unit 1 should be revoked and that certain measures should be taken to assure continued safe operation of Units 2 and 3. Therefore, your petition has been granted in part and denied in part.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. 20555 and in the local public document room at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Director's Decision
Under 10 CFR 2.206

MA-Z

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DD-80-5

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
HAROLD R. DENTON, DIRECTOR

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| In the Matter of | ) | |
| | ) | |
| CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. (Indian Point Unit Nos. 1 and 2) | ) | Docket Nos. 50-3 50-247 |
| | ) | |
| POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point Unit No. 3) | ) | Docket No. 50-286 |

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By petition dated September 17, 1979, the Union of Concerned Scientists (UCS) requested that the Nuclear Regulatory Commission (the Commission) revoke the provisional operating license for Indian Point Station Unit 1, order the licensee to submit a plan to decommission Unit 1, and suspend operation of Units 2 and 3 pending resolution of various safety-related issues. The UCS asks the Commission to hold a hearing on the matters raised in the petition as a basis for determining whether to permit resumed operation of Units 2 and 3. Consolidated Edison Company of New York (Consolidated Edison) holds the provisional operating license for Unit 1 and the operating license for Unit 2. The Power Authority of the State of New York (PASNY) holds the operating license for Unit 3. On October 26, 1979, the Commission formally referred the UCS' petition to the NRC Staff (the Staff) for treatment pursuant to 10 CFR 2.206. A notice that the petition was under consideration was published in the Federal Register, 44 FR 67251, on November 23, 1979.

Various persons have submitted responses to the UCS petition or have indicated their support of the petition. The two licensees each submitted responses, both dated September 28, 1979, to the UCS petition. The UCS replied to these two responses on October 25, 1979, with corrections dated October 30, 1979. The Commission has also received statements in support of the UCS petition on behalf of the Attorney General of the State of New York (November 16, 1979), from the Mid-Hudson Nuclear Opponents (November 27, 1979), from the New York Public Interest Research Group (January 3, 1980), from the Citizens Energy Council (January 4, 1980), from the Lead and Environmentally Aware Future (January 12, 1980), and from Women Opposed to Nuclear Technology (January 14, 1980)*. The Scientists and Engineers for Secure Energy, Inc., filed a statement opposing the UCS petition (January 29, 1980). Also, several members of Congress from New York and other members of the public have expressed interest in the UCS petition. At a meeting held on February 5, 1980, the Commission heard various organizations and members of the public express their views on the UCS petition and was briefed by the Staff on its proposed disposition of the petition.

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* These statements do not contain requests for relief or provide bases for relief that differ substantially from those found in the UCS petition. The staff has considered these statements in its review of the UCS petition. The New York Public Interest Research Group (NYPIRG), however, also cites in its statement potential dangers of theft of spent fuel and of a terrorist takeover of the Indian Point Station as a basis for suspending or revoking the Indian Point licenses. In the absence of facts which would substantiate these fears, NYPIRG has not provided a sufficient basis for the relief requested as required under 10 CFR 2.206(a). The staff continues to reexamine the compliance of these units with security regulations, and deficiencies so noted will be corrected. The licensses have made significant improvements in security as required by 10 CFR 73.55, which will provide adequate protection from such threats. In addition, the risks of accidents resulting from malevolent action will be reduced by the interim and long term action described herein. Some of these statements also cite concerns regarding the Ramapo fault, contamination of ground water and geology of the site. Concerning the Ramapo fault, the Staff, and Atomic Safety and Licensing Appeal Board have concluded that the fault is not a capable fault within the meaning of Appendix A to 10 CFR Part 100 of the Commission's regulations. The ACRS examined the site seismicity and did not disagree with these conclusions. The Indian Point 3 Safety Evaluation, dated September 21, 1973, considered potential contamination of ground water sources, the location of the Hudson River and the geology of the site and concludes that the site was acceptable.

The UCS gives four primary bases for requesting the revocation of the Unit 1 provisional operating license and the suspension of the Unit 2 and Unit 3 operating licenses:

(1) Unit 1, which has not operated since 1974, lacks safety features required to permit its resumed operation. However, the licensee has not pursued its application for a full term license or indicated that it intends to install necessary safety equipment, and therefore the provisional license for Unit 1 should be revoked and the facility decommissioned;

(2) The Indian Point Station is located in a densely populated area, which raises questions concerning the suitability of the site, the feasibility of evacuation of the area around the site, and the need for additional protective measures to assure safe operation of the Indian Point reactors;

(3) Unit 2 does not have some of the design features or equipment found in the subsequently licensed Unit 3; and

(4) Safety deficiencies and unresolved safety issues common to Units 2 and 3 require resolution before operation of the facilities is continued.

The Staff's evaluation and response to the UCS petition is contained in the remainder of this decision. As discussed herein, the Staff agrees that certain measures should be taken to assure continued safe operation of Units 2 and 3 and that the provisional operating license for Unit 1 should be revoked. Accordingly, the UCS petition is granted in part and denied in part.

I.

LICENSE REVOCATION AND DECOMMISSIONING UNIT 1

UCS asks (at pp. 10-13) that the Commission immediately revoke the Indian Point Station Unit 1 Provisional Operating License No. DPR-5 and order Consolidated Edison to present a plan for decontaminating and decommissioning the facility. The main thrust of UCS' complaint, with which the Staff essentially agrees, is that the pending application for conversion of License No. DPR-5 into a full-term operating license should not be permitted to continue in "regulatory limbo" and thereby result in an indefinite extension of License No. DPR-5.

Indian Point Station Unit 1 received License No. DPR-5 on March 26, 1962 under the authority of a since repealed portion of 10 CFR 50.57 [25 FR 8712 (1960), repealed, 35 FR 5317 (1970)], which provided for issuance of a provisional operating license as an interim step prior to issuance of a full-term operating license. Under 10 CFR 50.57, provisional operating licenses were issued for periods of 18 months, and extensions could be authorized for "good cause." After several extensions, License No. DPR-5 was set to expire on December 16, 1969. The licensee submitted, however, on November 10, 1969, an application to convert License No. DPR-5 to a full-term operating license. Under the terms of the Commission's regulations, the application had the effect of extending the Provisional Operating License No. DPR-5, until such time as the application "has been finally determined" [10 CFR 2.109*]. Because the application for the full-term

* This provision of the Commission's regulations reflects one of the procedural protections provided to licensees under the Administrative Procedure Act, specifically, the final sentence of Section 9(b) of the APA, 5 U.S.C. 558(c), which states: "When the licensee has made timely and sufficient application for a renewal or a new license in accordance with agency rules, a license with reference to an activity of a continuing nature does not expire until the application has been finally determined by the agency." The Staff agrees, however, that 10 CFR 2.109 should not be used to indefinitely extend an old license when the status of an application for a new or renewed license has remained essentially inactive for a long time.

license has not been "finally determined," License No. DPR-5 is not "deemed to have expired" as provided in 10 CFR 2.109.

Since October 1974, however, License No. DPR-5 has been an "operating" license in name only. Unit 1 has been in a shutdown condition since October 31, 1974, which was the expiration date of a variance [39 FR 29215 (1974)] granted to the licensee from the requirements of the Commission's "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors." On September 23, 1975, the Commission denied: (a) a request by the licensee for authorization which would have required another variance from the Interim Acceptance Criteria, (b) an exemption from the containment testing requirements of Appendix J to 10 CFR Part 50, and (c) extensions of time for compliance with two Commission Orders concerning other matters [40 FR 44895 (1975)]. There is presently no fuel in the Unit 1 reactor, and under the terms of License No. DPR-5 (Appendix A, Technical Specification 3.2.1), no fuel may be loaded into the reactor core or even moved into the reactor containment building without prior review and authorization by the Commission. Calculations have been made by the Staff and the licensee that show that the spent fuel now in the spent fuel pool has decayed sufficiently such that, in the event of a loss of water in the pool, this fuel can be air-cooled. Thus, there is no significant safety problem associated with the plant in its present defueled condition.

Since Unit 1 cannot meet current operational requirements and no plans exist for bringing it into compliance with current requirements, the operating provisions of License No. DPR-5 are not necessary. Accordingly, I have issued to Consolidated Edison the enclosed Order to Show Cause (Appendix A). The Order requires the licensee to show cause why the operating provisions of License No. DPR-5 should not be revoked and why the licensee should not submit a plan to decommission the facility. Thus, to that extent the UCS petition insofar as it concerns Unit 1 is granted.*

II.

INDIAN POINT UNITS 2 AND 3 AND POPULATION DENSITY

With regard to Indian Point Units 2 and 3, the petition alleges (at pp. 3, 6-10) that the consequences of a serious accident at the Indian Point

* The petition (at p. 23) asks that the Commission "immediately" revoke License No. DPR-5. Because the Commission must follow the provisions of section 9(b) of the APA in revoking any license under the Atomic Energy Act [sec. 186b. 42 U.S.C. 2236(b)], the Commission would have to find either that the licensee had wilfully committed (or omitted) some act for which a license could be revoked [see sec. 186 a.] or that the public health, safety or interest requires immediate revocation. No violations of the Commission's requirements are at issue here, and as noted in the text supra, no significant safety hazard is posed by the plant in its present condition. The Staff does not believe, therefore, that an adequate basis exists for ordering the immediate revocation of License No. DPR-5.

The net effect of the instant Order to Show Cause is the same as an immediately effective order revoking the license of an operating plant. If Indian Point Unit 1 were operating, the immediately effective order would suspend further operation of the facility during the proceeding on the order. In the actual case before the Commission, Indian Point Unit 1 is not operating and may not operate without the Commission's approval of exemptions from its regulations and changes to the license. In light of these facts, it is unnecessary to "immediately" revoke License No. DPR-5.

site because of a large surrounding population could be "enormous," and that, therefore, the Commission should determine the potential consequences of a "Class 9 accident," especially a core meltdown with breach of containment, as a basis for deciding whether these potential consequences are so severe as to render the Indian Point site unsuitable for a nuclear power plant. Each of the items identified in the petition pertaining to Indian Point Units 2 and 3 are addressed later in this decision. However, it is appropriate to first discuss separate efforts currently under way by the NRC Staff dealing with Indian Point Units 2 and 3 since it is believed that these efforts will adequately address the potential problems posed by the relatively high population density in the vicinity of the Indian Point site.

## NRC STAFF EFFORTS

Subsequent to the Three Mile Island accident, the Staff recognized the need to reassess the emergency preparedness plans and capabilities of all nuclear power plants. Because of their location in areas of high population density, the Indian Point Station Units 2 and 3 and Zion Station Units 1 and 2 (located north of Chicago, Illinois) facilities were recognized as plants for which additional measures might be necessary, including the possibility of a power reduction or plant shutdown.

An NRC Task Force has been formed to review Indian Point Units 2 and 3 and Zion Station Units 1 and 2. In addition the Staff, in conjunction with the Federal Emergency Management Agency (FEMA), is making emergency preparedness evaluations of these and other plants. These efforts, as they relate to the UCS petition, are discussed in detail below.

Emergency Preparedness Evaluations

On September 25 and 26, 1979 at meetings with both licensees, the Staff discussed its new criteria for developing emergency plans. These criteria were sent to all power reactor licensees in a letter dated October 10, 1979. On November 9, 1979, Consolidated Edison and PASNY submitted revised emergency plans in accordance with the new Staff criteria. On December 18, 1979, at a meeting held with the licensees, state and local officials, and members of the public, the Staff's review of these revised plans was discussed. The licensees were requested to resubmit their plans, revised to reflect Staff comments, within two months of the meeting. State and local officials have indicated they would cooperate with the licensees in developing these plans.

Until these revised plans are reviewed and accepted by the Staff, the licensees have put into effect emergency plans, submitted in March 1979, to conform with Regulatory Guide 1.101. We find that it is acceptable for the plants to continue operation while review of the revised plans of the licensees continues. The Commission, in the Proposed Rule on Emergency Planning published in the Federal Register [44 FR 75167, 75169 (December 19, 1979)] recognizes "that the increment of risk involved in operation of reactors over the prescribed times in the implementation of this rule [by January 1, 1981] does not constitute an unacceptable risk to the public health and safety." Similarly, the Staff does not believe that "the increment of risk" involved in operation while we are reviewing the licensees' plans during 1980 requires suspension of operation of Indian Point Station Units 2 and 3.

NRC Task Force

In addition to the in-depth review and development of the new emergency plans discussed above, an NRC Task Force has been designated to review two sites of operating nuclear power plants, Indian Point and Zion, that are located in areas of relatively high population density. The purpose of this Task Force is to review these facilities to determine what additional measures and/or design changes can and should be implemented that will further reduce the probability of a severe reactor accident and will reduce the consequences of such an accident by either reducing the amount of radioactive releases and/or by delaying any radioactive releases which would provide additional time for evacuation near the sites. The Task Force has evaluated certain interim measures that should be implemented by the licensees while the possible system design changes are being examined. Other measures will continue to be evaluated in the next few months. Some of the design changes being considered are a vented, filtered containment atmospheric release system, core retention devices, and hydrogen control.

Since design changes that may be decided upon will take one to two years to completely install, the Staff has identified, as part of the Task Force effort, a number of extraordinary interim measures that will be accomplished both by the licensees and by the Staff. These measures will significantly increase the level of safety at the Indian Point Station and thereby further reduce the probability and/or consequences of a severe reactor accident. By letters dated February 1, 1980, both licensees documented their commitment to implement these measures. I have formally confirmed this commitment by issuing Confirmatory Orders requiring this implementation at each of the two plants, Unit 2 and Unit 3. A copy of each of these Orders is provided as Appendices B and C to this Decision.

Included among those actions that are effective immediately by these two Confirmatory Orders are matters dealing with modes of operations, shift manning levels, enhanced training of operators, and special containment and low pressure-high pressure interface tests designed to add to the level of safety of operation of the facilities. Other requirements are to be implemented at various time intervals as specified in the Orders.

Those actions to be implemented by the Staff over and above those accomplished by the licensees include changes to the facility Technical Specifications to cause the Limited Conditions of Operation for safety-related systems to be at least as conservative as those in the Standard Technical Specifications for Westinghouse designed plants. In addition, enhanced Inspection and Enforcement presence will be established by providing a senior resident inspector for each operating Indian Point unit as well as a unit resident inspector.

Other Safety Considerations

In addition to the efforts described above, it should be pointed out that several compensating features already exist in the design of the Indian Point Station Units 2 and 3 which would limit the potential radiological consequences of a major accident. These include:

1. A containment weld channel and weld channel pressurization system: All containment liner welds are enclosed by continuous linear channels welded to the liner to form a redundant seal at the joints of liner plates. Those channels which cover joints not buried in concrete are pressurized with air to a pressure exceeding calculated containment peak pressure. This eliminates leakage at liner plate joints.

2. A penetration pressurization system: In addition to the normal pressurization of electrical penetrations (with dry nitrogen), mechanical penetrations are pressurized with air to a pressure above calculated containment peak pressure. This eliminates leakage through penetration assemblies.

3. An isolation valve seal water system: Those double isolation valves, normally closed on a containment isolation signal, in water and small air systems, have the area between valves filled (if needed) and maintained in a filled condition at a pressure exceeding calculated containment design pressure by this system. This eliminates any leakage of containment atmosphere via an open (or ruptured) line through the redundant isolation valves.

4. Extra containment fan cooler capacity: Each containment has five fan cooler units, three of which are required for post accident containment cooling. The added capacity provides assurance of system availability.

5. Post-LOCA hydrogen control: Each unit has both recombiner and post-LOCA containment purge capability. The recombiner capability was added to provide additional conservatism.

6. A third auxiliary feedwater pump: Each unit has three auxiliary feedwater pumps. Two of these are 100% capacity motor driven pumps and the third is a 200% capacity steam turbine driven pump. All three pumps are intertied through lines and valves designed for an active or passive failure. This extra capacity over a 2-100% capacity pump configuration provides added assurance of system availability.

7. Containment atmosphere radioactivity removal (cleanup) has been provided. Each fan cooler unit is equipped with HEPA and charcoal filters for post-accident particulate and iodine radioisotope removal by entrapment.

8. Confirmatory Emergency Safeguards Features (ESF) actuation signals are sent to power operated valves which are not required to change position. This ensures that, if a valve had inadvertently been placed in an incorrect position, it would move to the correct position upon ESF actuation. This has been applied to critical safety system valves.

In addition, each unit has additional margin in service water and component cooling water capacity and availability. They have auxiliary building air filtration (cleanup) systems and closed valve leak off systems to reduce offsite exposure due to valve stem leakage. They also have redundant electrical heat tracing on vital borated systems.

Thus, considering these existing engineered safety features, the emergency plans already in effect, and the extraordinary interim measures identified in the Confirmatory Orders, I have determined that Indian Point Station Units 2 and 3 are suitable for continued operation pending completion of the design reviews being performed by the NRC Task Force and pending completion of the Staff's review of the revised emergency plans.

## III.

## OTHER MATTERS IDENTIFIED IN THE PETITION

### Differences in Design Between Unit 2 and Unit 3

As a basis for requesting the suspension of operation of Unit 2, the UCS alleges (at pp. 13-17) that the designs of Unit 2 and Unit 3 differ in ways that have a "significant effect" on the risk to public health and safety created by operation of each unit. Therefore, UCS argues, the Commission should immediately backfit Unit 2 to incorporate changes made to Unit 3 as a result of the Staff's review of that unit. The UCS also requests the Staff to identify all design changes made "voluntarily" to Unit 3 to determine whether these changes should be implemented at Unit 2. The UCS identifies three features which the UCS believes require immediate action: diesel generator buildings, battery system and auxiliary feedwater system.

The Confirmatory Orders (Appendices B and C) require that within 90 days the licensees jointly identify and review the significant differences between Unit 2 and Unit 3, and that they evaluate these differences in light of present regulatory standards and requirements. The licensees are required to provide a justification for the current design, or provide design change recommendations.

In addition, it should be noted that numerous changes have already been made to Unit 2 as a result of the licensee's review of Unit 3. During the licensing of Indian Point Unit 3, the Staff and the licensee (at that time Consolidated Edison was the licensee for both Indian Point Units 2 and 3) did re-evaluate Indian Point Unit 2. As a result of this re-evaluation, described in a letter dated September 4, 1976, transmitting Amendment No. 20 from Robert W. Reid, NRC, to William J. Cahill, the following changes were made to Unit 2:

1. A second independent and redundant Safety Injection (SI) Block Switch was added.

2. Separate annunciation devices were installed which alarm when either train of Engineered Safety features logic has been bypassed.

3. A second independent pressure transmitter was installed to provide a separate, independent interlock signal to the Residual Heat Removal (RHR) suction valves 730 and 731.

4. The electrical interlock between SI valves 888A and 888B and RHR valves 730 and 731 was changed such that valve 730 was interlocked with valve 888A and valve 731 was interlocked with valve 888B.

5. Contacts, which open upon safety injection actuation, were added in series with the following switches or interposing relay contacts:

   a. Switch 3
      "43/RS-3" trip to each RHR pump
   b. Switch 6
      "43/RS-6" open signal to valves 888A and B
      "43/RS-6" close signal to valves 746 and 747
   c. Switch 7
      "43/RS-7" trip to each SI pump

6. Miniflow bypass valves 743 and 1870 for the RHR pumps were made passive by having their electric power physically disconnected and locked in the open position.

7. Two circuit interrupting devices were added between the automatic transfer device and each DC bus. (See subsequent discussion on automatic transfer devices and battery system.)

In addition to these modifications resulting from a comparison to Indian Point 3, other reviews resulted in further backfitting at Indian Point Unit 2. Some significant items include security improvements to meet 10 CFR 73.55, fire protection (described in our SER dated January 31, 1979 supporting Amendment No. 46), installation of "J-tubes" to prevent feedwater hammer, modifying or relocating valves and electrical equipment inside containment that would have been submerged following a loss-of-coolant accident, modifications to eliminate single failures of ECCS, modifications to preclude overpressure events, and modifications to meet the TMI-2 lessons learned requirements.

Nevertheless, as indicated above, the licensee is required to perform a review and justify any significant differences that currently exist between the two units, because all significant differences may not have been evaluated during the previous reviews.

The petition cites three specific examples of alleged safety significant design differences between Indian Point 2 and 3. These are the diesel generator building, the battery system and the auxiliary feedwater system. Each of these is discussed below.

Diesel Generator Building

The Staff's fire protection review of Indian Point Unit 2 required that significant changes be made to the diesel generator building. As stated in our January 31, 1979 Safety Evaluation Report (SER), the licensee will erect shields between the diesel generator units, provide one-hour fire proofing on the building structure, and install backflow prevention check valves on drain lines. The fire proofing on the building structure was completed during the summer 1979 refueling outage, and the other modifications will be completed by the end of the next refueling outage, presently scheduled for December 1980.

In addition, fire protection is provided by an automatic sprinkler system in the area, heat detectors that alarm in the control room, and fire hoses from fire hydrants near the area. The licensee has also implemented administrative procedures to prevent conditions that could lead to a fire, such as housekeeping inspections and use of protective blankets and fire watches during welding operations. A trained fire brigade onsite for all shifts has also been established.

Furthermore, as stated in the fire protection SER, the capability to attain safe shutdown (within 72 hours) and maintain safe hot shutdown independent of the diesel generators or offsite power will be provided by the end of the next refueling outage.

With respect to tornadoes, the location of the Indian Point Unit 2 diesel generator building makes it less susceptible to high winds than the Indian Point Unit 3 diesel generator building. Page 34 of the Staff's "Safety Evaluation of the Indian Point Nuclear Generating Unit No. 2," dated November 16, 1970, states: "Some natural protection from high winds is afforded the control room building and diesel generator building since they are protected by the turbine building to the west, the Indian Point Unit 1 turbine building, superheater building and containment to the south, the rising hillside to the east, and the containment and rising hillside to the north." The conclusion in that report "that Indian Point Unit 2 is adequately protected against high winds," is still valid.

Finally, there are presently available, and separately located, three gas turbine generators, at least one of which is required to be operable (Amendment No. 60, dated January 28, 1980) to place the reactor in a safe shutdown condition in the event that all three diesel generators and offsite power were lost.

Due to the protective features afforded the diesel generator building and due to the availability of other power sources, the Staff has concluded that the diesel generator building is acceptable pending completion of the above described modifications.

Battery System

The UCS alleges that the battery system for Indian Point Unit 2 is inadequate because the system contains only two batteries and relies on automatic transfer switching.

There are seven automatic transfer circuits used with engineered safeguards. Three automatic transfer circuits provide redundant 125V DC control power to the three diesel generators. The remaining four transfer circuits provide redundant power to the 480V diesel generator switchgear. Each transfer device receives its 125V DC power from the same two emergency battery buses. Two circuit interrupting devices between the auto transfer device and each DC bus have been provided. The Staff has verified that no single failure in the transfer device circuitry would cause the loss of either DC bus. Although it is possible to connect redundant power sources in parallel considering an undetected failure, two separate short circuits to ground (or a line to line short) and the failure to function of four overcurrent protection devices would be required to compromise redundant DC buses.

Ground detectors are used as an integral part of the Westinghouse battery chargers. If a ground were to be present on a DC bus, a ground indicating light would go out and a "battery charger trouble" alarm would annunciate in the central control room. The circuit grounding problem would thus be promptly detected, isolated, and corrected. Also, the licensee has incorporated a test procedure in its periodic battery testing program to assure operability of the ground detection system. Therefore, the design of these automatic transfer circuits, with the above periodic testing, meets the single failure criterion. On that basis, the Staff has concluded that a single failure in this system would not lead to a meltdown as alleged. Nevertheless, the Staff is re-evaluating the acceptability of the automatic transfer feature of this system. Furthermore, during the fall 1978 refueling outage, the battery system was upgraded by the installation of two additional batteries to provide power for two channels of instrumentation (bringing the total to four batteries for Indian Point Unit 2). The modification is described in the March 1, 1979 letter from William J. Cahill, Jr. to Boyce Grier, Director of NRC's Region I Office.

Auxiliary Feedwater System

The third specific item allegedly requiring backfitting is the auxiliary feedwater (AFW) system. A thorough review of the Indian Point Unit 2 AFW system was conducted by the Staff. The results were transmitted to the licensee on November 7, 1979. This NRC letter identified additional requirements for the AFW system. Consolidated Edison in its response dated December 19, 1979 proposed the following modifications:

1. Revise the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable.

2. Develop emergency procedures for transferring to the alternate source of AFW supply.

3. Make the automatic start AFW system signals and associated circuitry and AFW flow indication safety-grade. (This is being done in conjunction with the NRC TMI-2 Lessons Learned Task Force Recommendations 2.1.7.a and 2.1.7.b.)

4. Develop procedures to assure AFW system function in the event of abnormal failure of the pneumatic operated AFW flow control or steam supply valves.

5. Install a redundant level indication and low level alarm system on the condensate storage tank with annunciation in the control room.

6. Install a redundant flow path, with manual redundant valves, in parallel to the single flow path from the condensate storage tank.

7. Evaluate the capability of the present AFW system design to withstand internally generated missiles, and make any modifications deemed necessary.

The procedures identified in items (2) and (4) have already been put into effect and the revision to the Technical Specifications proposed in item (1) has already been issued in Amendment No. 60, dated January 28, 1980.

The hardware modifications identified in items (3), (5), (6) and (7) will be completed on an expedited basis as required by the Confirmatory Order.

The petition specifically alleges that a break in the steam pipe to the turbine-driven AFW pump could result in a total loss of AFW because the motor-driven pumps are located in the same room as the turbine-driven pump. As a result of studies of high energy line failures and flooding of areas containing safety-related components, certain plant modifications were made to protect the AFW system from the effects of a break in the steam pipe to the turbine-driven AFW pump. These include: (1) installation of isolation valves in the steam pipe, external to the room, that will close upon sensing high temperature in the room; and (2) modifications made to the doors to assure adequate drainage.

We conclude that the new procedures and Technical Specifications, in addition to modifications completed and scheduled to be completed on the auxiliary feedwater system within the time indicated above, are adequate to allow continued operation of the Indian Point Unit 2.

Other Safety Deficiencies Identified in the Petition

In addition to those items for Indian Point Unit 2, the petition alleges that there are other safety deficiencies, common to both Indian Point Units 2 and 3, that require suspension of operation of both units pending their resolution.

Cable Spreading and Fire Protection Systems

Paragraphs 50 through 54 of the petition concern cable separation and fire protection systems for those areas where fires could affect redundant divisions of shutdown systems. The UCS previously raised these issues in its petition to the Commission concerning the adequacy of fire protection on an overall basis at nuclear power plants. These items have been previously addressed generically in information provided by the Staff to the Commission to assist its evaluation of the UCS petitions of November 1977 and May 1978. The UCS petition on Indian Point (paragraphs 50 through 54) does not contain any information relative to fire protection which indicates the need for immediate action at Indian Point beyond any actions that may result from the Commission's final determination on the November 1977 and May 1978 petitions.

Nevertheless, many changes have been made, and are scheduled to be made, related to fire protection. These are discussed in detail in our Fire Protection Safety Evaluation Reports, January 31, 1979 for Indian Point Unit 2 and March 6, 1979 for Indian Point Unit 3. We find no basis to alter our conclusion that the schedule for completion of the remaining fire protection issues is acceptable and does not require a plant shutdown pending their completion.

Unresolved Safety Issues

The petition also refers to the 133 "unresolved safety issues" identified in an NRC Report to Congress. The items are identified in NUREG-0410 "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plant," dated January 31, 1978, and cover a variety of topics. Only some are related to safety; others are related to environmental matters and improving the regulatory process. We reported in NUREG-0510, "Report to Congress by the NRC Staff on Identifying Unresolved Safety Issues," dated January 31, 1979, that only 22 of these 133 generic tasks were "unresolved safety issues."

Furthermore, with respect to those tasks of safety significance, we discussed generically in NUREG-0510 the NRC's basis for permitting a plant to continue to operate with an "unresolved" safety issue. The bases for such a determination are (1) the issue does not apply, or has been resolved, for the plant under consideration; (2) interim measures assuring adequate safety of operation are being required at the plant pending final resolution of the issue; (3) resolution of the issue can be reasonably expected before the plant under consideration begins (returns to) operation, or (4) the likelihood of occurrence and/or the safety consequences of a scenario dealing with the issue is small. The Staff has specifically re-examined these issues for Indian Point Units 2 and 3 and has decided that continued plant operation is acceptable for the above reasons for each of the outstanding issues. Furthermore, the Staff is making a concerted effort to accelerate resolution of outstanding generic and plant specific actions pertinent to Units 2 and 3.

The UCS notes (at p. 20) that there has been "no systematic evaluation of the need to upgrade Indian Point to account for important safety lessons learned." The Commission, as reflected in letters dated December 17, 1979 and January 3, 1980 from Chairman Ahearne to Representative Morris Udall, agrees that the NRC should undertake a comprehensive program for systematically reevaluating the safety of <u>all</u> currently operating plants. Copies of those letters are attached as Appendices D and E to this determination. In particular, the December 17, 1979 letter provided comments on an amendment to H.R. 2608 offered by Representative Bingham. The letter states:

> "...two years ago the Commission undertook a reevaluation on a limited basis with respect to all of the older operating plants. We believe a variation of this Systematic Evaluation Program should be developed for application to all operating plants. Such a program should also address generic safety issues... It will take several months for the NRC staff to develop and propose, and for the Commission to approve, this systematic program for evaluating the safety of all operating plants. It most likely will include some elements of the ongoing Systematic Evaluation Program, in which evaluations are being made by the NRC staff of the design of the older plants with regard to some 130 safety 'topics'."

In addition to its general allegations concerning safety issues common to Units 2 and 3, the UCS specifically alleges that three unacceptable safety problems exist related to post-accident monitoring, aging of equipment, and asymmetric loads on the reactor.

Post-Accident Monitoring

The petition alleges that the Three Mile Island accident demonstrated the inadequacy of the post-accident monitoring. First of all, it must be recognized that the designs of instrumentation for Indian Point Unit 2 and 3 are different from Three Mile Island (TMI) Units 1 and 2 because the plants were designed by different nuclear steam suppliers. For this reason, some equipment (e.g., pressurizer level) may have a safety function in one plant and not in another. The pressurizer instrumentation for Indian Point Units 2 and 3 has a safety function and is already Class 1E whereas TMI's instrumentation did not have a safety function and was not class 1E. Because the pressurizer level measurement system in TMI was not required for safety, it was not protected from containment flooding nor was it reviewed for its capability to survive an accident or post accident environment.

We know of no Class 1E instrumentation at TMI that has failed to provide the required accuracy during or after the TMI accident. The fact that pressurizer level was needed at TMI (and survived the accident environment, even though it was not environmentally qualified for an adequate period) contradicts the petitioner's argument of inability to monitor the parameters, the range and accuracy of the instrumentation, ability of the instrumentation to survive the accident and post-accident environment. We do, however, acknowledge that by Bulletins and Orders and Lessons Learned activities we have required specific instrumentation improvements on a specified schedule. The licensees have met our requirements in this regard.

Post-accident monitoring has already been improved as part of the implementation of the TMI-2 Lessons Learned Short Term Requirements. The following modifications have been made on Unit Nos. 2 and 3.

1. A reactor coolant saturation meter (subcooling meter) to provide on-line indication of coolant saturation condition was provided. This will aid the operator in recognizing inadequate core cooling.

2. An acoustic monitoring system for positive pressurizer relief safety valve position indication was installed.

3. A plan has been established for an onsite radiological and chemical analysis facility with the capability to provide, within one hour of obtaining the sample, quantification of certain isotopes that are indicators of the degree of core damage, hydrogen levels in the containment atmosphere, and dissolved gases and boron concentration in liquids.

The staff believes that appropriate action to upgrade instrumentation has been identified and is being implemented independent of this petition. The petition alleges that there is no way to directly measure the water level or temperature in the core after an accident. An adequate indication of core submergence is available from the pressurizer level measurement systems as long as the reactor coolant system is subcooled. (This has been demonstrated graphically by the TMI-2 accident.) As previously mentioned, both plants have installed subcooling meters to comply with our Short Term Lessons Learned requirements. The Staff therefore rejects the petitioner's allegations that the present lack of a direct measure of core water level is a safety deficiency since an acceptable alternate means of measurement is available.

With regard to core temperature measurements, the Staff maintains that measurement of hot and cold leg reactor coolant temperatures is sufficient to demonstrate that adequate temperature control is being exercised as long as adequate coolant circulation is maintained through the core. Core exit thermocouples are provided in Indian Point Units 2 and 3, which provide temperature indication directly adjacent to the core.

The petition alleges that the only temperature measurements at TMI-2 were from non-safety grade equipment, some of which "luckily" survived the accident. Other temperature measurements were available at Three Mile Island but were meaningless until coolant flow was established because the parameters of interest involved heat transfer from the core. The only sensors available in the circulation path (inside of the reactor vessel) were the core exit thermocouples. These sensors are not Class 1E and are not required for any event in which adequate reactor coolant flow is maintained. As the TMI accident proved, and our survey later confirmed, the type of thermocouples used are inherently capable of surviving events such as TMI to the extent necessary to protect public health and safety. The number and types of temperature measurement systems in pressurized water reactors are similar from plant to plant.

In addition to the instrumentation added as part of the Lessons Learned requirements, and instrumentation that was already in place, the following activities will take place during 1980:

1. Both licensees are part of the Westinghouse Owner's Group that is performing analyses to determine if additional instrumentation is necessary to provide a better indication of inadequate core cooling.
2. The existing auxiliary feedwater flow indication will be upgraded to safety grade.
3. Extended range noble gas effluent monitors will be installed.
4. The capability for effluent monitoring of radioiodines will be established.
5. Extended range in-containment radiation level monitors will be installed.
6. Containment pressure indicators capable of measuring containment pressures up to three times the design accident pressure will be installed.
7. A continuous indication of hydrogen concentration in the containment will be provided.
8. Improvements will be made to the instrumentation for measuring containment water level.

The above modifications, and the schedule for implementing them, are consistent with our Lessons Learned requirements. We, therefore, conclude that immediate shutdown of the two facilities is not necessary to upgrade post-accident instrumentation.

Equipment Aging

The staff acknowledges that new equipment may have been used in the original equipment qualification testing for Indian Point Units 2 and 3, and that no systematic effort was made to determine the length of time in service during which the results would remain valid. In order to assure that this aspect of equipment qualification is adequately addressed, the staff has included consideration of the potential effects of aging in its current program to reevaluate the adequacy of equipment qualification in all operating reactors. This reevaluation is being conducted in conjunction with our review of the licensees' responses to IE Bulletin 79-01, "Environmental Qualification of Class 1E Equipment".

The licensees' responses of June 13, 1979 to IE Bulletin 79-01 will be evaluated in accordance with a set of screening guidelines set forth in a Staff document entitled, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" which was transmitted with IE Bulletin 79-01B, dated January 14, 1980. The Bulletin requires additional information and evaluations from the licensees. Under these guidelines a specific qualified life should be established for equipment using materials that have been identified as being susceptible to significant degradation due to thermal and radiation aging. A list of materials which may be found in nuclear power plants along with an indication of the material susceptability to thermal and radiation aging is provided in an Appendix to the guidelines. In addition, under the guidelines, ongoing programs should be in existence at the plant to review surveillance and maintenance records to assure that equipment which is exhibiting age related degradation will be identified and replaced as necessary.

We believe that the program outlined above provides reasonable assurance that equipment subject to significant degradation due to aging will be identified and that maintenance or replacement schedules will be adjusted accordingly. The Staff, additionally, is acceleratng its evaluation of the adequacy of the equipment qualification program at the Indian Point plants. In the interim, the margins that exist in the equipment design provide reasonable assurance that equipment will function as required in the event of a design basis accident.

Asymmetric LOCA Loads

Another specific area discussed in the UCS petition deals with asymmetric loads from a postulated accident on the reactor. A generic study of the asymmetric loss of coolant accident (LOCA) loads problems was initiated by the Staff in 1977 to both gain a better understanding of this problem and to develop criteria for plant specific evaluations. This generic study, Task Action Plan A-2, described in NUREG-0510, was essentially completed in late 1979 and is expected to be published as a NUREG in February 1980.

Plant specific evaluations for the Indian Point 2 and 3 plants have been submitted to the Staff and are currently being reviewed against criteria derived from the Staff's generic study. The Staff's review is expected to be completed early in 1980. Until our review is complete, and modifications to the facilities are made, we have concluded that there is reasonable assurance that continued operation, pending completion of this task, does not constitute an undue risk to the health and safety of the public for the following reasons.

As discussed below, the likelihood of occurrence of an initiating event of sufficient magnitude to seriously challenge the structural adequacy of the vessel support members or other structures is low. The disruptive failure of a reactor vessel itself has been estimated to lie between $10^{-6}$ and $10^{-7}$ per reactor year, so low that it is not considered as a design basis event. The rupture probability of pipes is estimated to be higher. The data base used by WASH-1400* indicates a median value of $10^{-4}$ for LOCA initiating ruptures per plant-year for all pipe sizes 6" and greater (with a lower and upper bound of $10^{-5}$ and $10^{-3}$, respectively). We believe that considering the large size of the pipe in question (up to 50" O.D. and 4-1/8" thick), a median value nearer $10^{-5}$ than $10^{-4}$ is more appropriate using the same data base. In addition, the quality control of the piping used in nuclear power plants is somewhat better than that of conventional piping, the piping whose data was used in most probability evaluations.

Because (1) the break of primary concern must be large and is of low probability, (2) only certain break locations lead to high loads, and (3) these welds are currently subject to preservice and inservice inspection by volumetric and surface techniques in accordance with ASME Code Section XI, we conclude that the probability failure of a pipe system or other structures is acceptably small and that reactor operation can continue while this matter is being resolved.

* WASH-1400 was only used to support the Staff's engineering judgment, as stated in SECY 79-106 to the Commissioners.

IV.

## CONCLUSION

The petition alleges that Indian Point Units 2 and 3 are "relics of the past" and the "NRC has marched resolutely 'eyes front', not applying the lessons learned about safety to Indian Point."

This is not so. Both plants have been significantly modified to meet NRC safety and security requirements. The safety modifications are too numerous to list, but many may be found in the correspondence between the NRC and the licensees that is available for public inspection in the NRC's Public Document Rooms and that includes the following documents:

1. TMI-2 Lessons Learned: NRC letters dated September 17, 1979 and October 30, 1979; Consolidated Edison letters dated October 17, 1979, November 20, 1979, December 7, 1979, December 17, 1979 and December 31, 1979; and Power Authority of the State of New York (PASNY) letters dated October 22, 1979, November 21, 1979, December 4, 1979, December 10, 1979, December 17, 1979 and January 8, 1980.

2. Fire Protection: NRC letters dated January 31, 1979 for Unit 2 transmitting Amendment No. 46, and March 6, 1979 for Unit 3 transmitting Amendment No. 24.

3. Overpressure Protection: Consolidated Edison letters dated February 28, 1977, April 5, 1977, August 9, 1977, September 20, 1979 and December 5, 1977.

In addition, the NRC Task Force described herein will determine what design changes should be made to further reduce the probability and/or consequences of a severe reactor accident. Until these changes can be implemented, the extraordinary interim measures identified in the attached Confirmatory Orders (Appendices B and C) will provide additional assurance of safe operation of these facilities.

Because of the interim measures imposed by the Confirmatory Orders and in light of the discussion in this decision of the safety issues raised by the UCS, I have determined not to order the shutdown of Indian Point Units 2 and 3. For these same reasons I have not recommended to the Commission that it institute a hearing on all of the matters touched upon in the UCS petition.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. 20555 and in the local public document room at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601. Additionally, a copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

Harold R. Denton
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
Appendix A: Order to Show Cause (Unit 1)
Appendix B: Confirmatory Order (Unit 2)
Appendix C: Confirmatory Order (Unit 3)
Appendix D: Letter to Representative Udall (12/17/79)
Appendix E: Letter to Representative Udall (01/03/80)

Dated this 11th day of
February, 1980.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of )
)
CONSOLIDATED EDISON COMPANY ) Docket No. 50-3
OF NEW YORK, INC. )
(Indian Point Station, Unit No. 1) )

ORDER TO SHOW CAUSE

I.

The Consolidated Edison Company of New York, Inc. (the Licensee) is the holder of Provisional Operating License No. DPR-5 (the license), which was issued on March 26, 1962 to authorize operation of Indian Point Station, Unit No. 1, located in Westchester County, New York. License No. DPR-5 was issued as a provisional operating license, and has continued in effect since 1969 under a timely application for a full-term operating license.

II.

Indian Point Station, Unit No. 1, received a provisional operating license on March 26, 1962 under the authority of a since repealed portion of 10 CFR 50.57 [25 FR 8712 (1960), repealed, 35 FR 5317 (1970)], which provided for issuance of a provisional operating license as an interim step prior to issuance of a full-term operating license. Provisional operating licenses were issued for periods of 18 months, and extensions could be authorized for "good cause." After several extensions, License

No. DPR-5 was set to expire on December 16, 1969. The licensee submitted, however, on November 10, 1969, an application to convert License No. DPR-5 to a full-term operating license. Under the terms of the Commission's regulations, the application had the effect of extending the provisional operating license, No. DPR-5, until such time as the application "has been finally determined" [10 CFR 2.109*]. Because the application for the full-term license has not been "finally determined," License No. DPR-5 is not "deemed to have expired" as provided in 10 CFR 2.109.

Since October 1974, however, License No. DPR-5 has been an "operating" license in name only. Unit 1 has been in a shutdown condition since October 31, 1974, which was the expiration date of a variance [39 FR 29215 (1974)] granted to the licensee from the requirements of the Commission's "Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Power Reactors." On September 23, 1975, the Commission denied: (a) a request by the licensee for authorization which would have required another variance from the Interim Acceptance Criteria, (b) an exemption from the containment testing requirements of Appendix J to 10 CFR Part 50, (c) and extensions of time for compliance with two Commission Orders concerning other matters [40 FR 44895 (1975)]. There is presently no fuel in the Unit 1 reactor, and under the terms of License No. DPR-5 (Appendix A, Technical Specification 3.2.1), no fuel may be loaded into the reactor core or moved into the reactor containment building without prior

* This provision of the Commission's regulations reflects one of the procedural protections provided to licensees under the Administrative Procedure Act, specifically, the final sentence of Section 9(b) of the APA, 5 U.S.C. 558(c).

review and authorization by the Commission. Calculations have been made by the NRC Staff (the Staff) and the licensee that show that the spent fuel now in the spent fuel pool has decayed sufficiently such that, in the event of a loss of water in the pool, this fuel can be air-cooled. Thus, there is no significant safety problem associated with the plant in its present defueled condition.

By letter dated September 23, 1976, the Staff noted that the licensee had not met Staff requirements in other areas, including containment isolation, reactor protection system and seismic design, and concluded that "the design of Indian Point 1 has thus become deficient in a number of respects".

By letter dated July 16, 1976, the licensee submitted an application for a license amendment to reflect the defueled, non-operating status of the reactor. The amendment was issued April 14, 1977 and prohibits the licensee, as indicated above, from loading fuel in the reactor, or moving fuel into containment, without NRC authorization. In the letter accompanying this amendment, the licensee was reminded "that our review for the restart of Indian Point Unit 1 would include all applicable issues which have arisen since the shutdown of Indian Point Unit 1."

Since Unit 1 does not meet current operational requirements and no plans exist for bringing it into compliance with current requirements, the useful life of Unit 1 as an operating nuclear power reactor is effectively at an end.

Therefore, the Staff intends to revoke the operating authority provided in License No. DPR-5. Accordingly, it is appropriate for the Licensee to submit a plan to decommission Unit 1 that would address, among other things, the extent to which the Licensee would dismantle the facility. In this regard, the Staff brings to the Licensee's attention the provisions of 10 CFR 50.82 and the guidance contained in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors."

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Part 2 and 50, IT IS HEREBY ORDERED THAT the Licensee show cause, in the manner hereinafter provided, why (1) the operating authority provided in Provisional Operating License No. DPR-5 should not be revoked; and (2) the Licensee should not submit pursuant to 10 CFR 50.82. a plan within 120 days of this Order to decommission Indian Point Station, Unit No. 1.

IV.

The Licensee may within thirty days of the date of this Order file a written answer to the Order under oath or affirmation. Within the same time, the Licensee or any person who has an interest affected by this Order may request a hearing on the Order. Any request for a hearing shall be addressed to Harold R. Denton, Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. If a hearing is requested by the Licensee or a person who has an interest affected by this Order, the Commission will issue an Order designating the time and place of hearing. Upon failure of the Licensee to file an answer within the time specified, the Commission will, without further notice, issue an order revoking the operating authority provided in License No. DPR-5 and requiring the Licensee to submit pursuant to 10 CFR 50.82 a decommissioning plan for Indian Point Station, Unit No. 1.

If a hearing is held, the issue to be considered at such hearing shall be:

Whether, on the basis of the facts stated in Section II of this Order, the operating authority of Provisional Operating License No. DPR-5 should be revoked and the Licensee required to submit pursuant to 10 CFR 50.82 a decommissioning plan for Indian Point Station, Unit No. 1.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 11th day of February, 1980.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of )
)
CONSOLIDATED EDISON COMPANY ) Docket No. 50-247
OF NEW YORK, INC. )
(Indian Point Station, Unit No. 2) )

CONFIRMATORY ORDER

I.

The Consolidated Edison Company of New York, Inc. (the Licensee) is the holder of Operating License No. DPR-26 (the license) which authorizes operation of the Indian Point Unit No. 2, located in Westchester County, New York, at steady state reactor core power levels not in excess of 2758 megawatts thermal (rated power).

II.

Due to the relatively high population density surrounding the Indian Point site as compared to other nuclear power plant sites, the Indian Point site is believed to present a disproportionately high contribution to the total societal risk from reactor accidents. The NRC Staff (the Staff) has currently under way two separate efforts to address the potential problems posed by this relatively high population density. One of the efforts involves the development, revision, and review of emergency plans. This effort is scheduled to be completed by January 1, 1981.

The other effort is a review of the Indian Point facilities to determine what additional procedural measures and/or design changes can and should be implemented that will further reduce the probability of a severe reactor accident and/or to reduce the consequences of such an accident. Since design changes that may be decided upon will take one to two years to completely install, the Staff has identified a number of extraordinary interim measures that should be accomplished both by the licensees and the Staff. These measures will significantly increase the level of safety at the Indian Point Station and thereby further reduce the probability of a severe reactor accident.

Included among these actions are matters dealing with modes of operations, shift manning levels, enhanced training of operators, and special containment and low pressure interface tests designed to add to the level of safety of operation of the facility. All requirements shall be implemented at the time intervals specified in this Order.

The Licensee, in a letter dated February 1, 1980, has agreed to undertake the actions listed in Appendix A to this Order. It is desirable to confirm the Licensee's commitment by Order.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT:

The licensee perform the actions stated in Appendix A to this Order. The aforementioned actions shall be performed in accordance with the schedule set forth in Appendix A or, in the alternative, the licensee shall place and maintain its facility in a cold shutdown condition within 48 hours pending completion of those actions.

IV.

Any person who has an interest affected by this Order may request a hearing within twenty (20) days of the date of the Order. Any request for a hearing will not stay the effectiveness of this Order. Any request for a hearing shall be addressed to Harold R. Denton, Director, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. If a hearing is requested requested by a person who has an interest affected by this Order, the Commission will issue an Order designating the time and place of any such hearing.

In light of the licensee's expressed willingness to undertake the actions ordered, if a hearing is held concerning this Order, the issue to be considered at the hearing shall be:

> Whether the licensee should perform the actions in Appendix A to this Order in accordance with the schedule stated therein.

Operation of the facility on terms consistent with this Order is not stayed by the pendency of any proceedings on the Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: February 11, 1980
Bethesda, Maryland

## APPENDIX A

A. The licensee shall:

1. Maintain reactor power level as necessary such that calculated fuel peak clad temperature does not exceed 2000°F under large break LOCA conditions.

2. Revise plant operating procedures as necessary to require a base load mode type of operation only, without load following.

3. Conduct a low presssure gross leak test of containment prior to any start up from cold shutdown conditions. If other means can be found to verify containment integrity, the licensee may propose such procedures to the Commission for its review and approval.

4. Maintain at least two senior reactor operators (SROs), one of whom may be the shift supervisor, in the control room at all times during power operations or hot shutdown, except that the shift supervisor shall be allowed to leave the immediate vicinity of the control room as duties may require, provided he is available to respond to an emergency by returning to the control room within ten minutes. The shift or watch supervisor's office is considered part of the control room.

5. Conduct testing to assure that the LPI/RHR check valves are in fact installed correctly and functioning as pressure isolation barriers when the plant is at pressure and producing power. Verification of valve operability shall be performed prior to plant restart if shutdown at the time of issuance of the Order and thereafter whenever RCS pressure has decreased to within 100 psig of RHR system design pressure.

6. Submit not later than March 1, 1980 the results of a review of possible permanent plant modifications and procedures to further reduce the potential of a severe reactor accident and resultant radiation releases.

7. Require that all reactor operators and senior reactor operators conduct simulator training and in-plant walk through of the following emergency procedures. The in-plant walk-throughs shall be completed prior to the next reactor startup following issuance of the Order, or within thirty days of the date of issuance, whichever occurs first. Those reactor operators and senior reactor operators who have not received simulator training within the past three months on these items shall be given such simulator training within 60 days of the date of the Order:

a. Plant or reactor startups to include a range wherein reactivity feedback from nuclear heat addition is noticeable and heat up rate is established

b. Manual control of steam generator level and/or feedwater during startup and shutdown

c. Any significant (10%) power change using manual rod control

d. Loss of Coolant

   (i) including significant PWR steam generator leaks

   (ii) inside and outside containment

   (iii) large and small, including leak rate determination

   (iv) saturated reactor coolant response (PWR)

e. Loss of core coolant flow/natural circulation

f. Loss of all feedwater (normal and emergency)

g. Station blackout

h. Anticipated Transients Without Scram (ATWS)

i. Stuck open relief valve on secondary side

j. Intersystem LOCA

B. The licensee shall implement the following measures within 30 days of the date of the Order:

   1. A vendor representative will be stationed on site for engineering consultation at Indian Point Unit 2 and Unit 3 on plant operations and maintenance to increase plant safety. The representative shall be from the NSSS vendor, architect/engineering or start up engineering firm.

   2. To ensure control room habitability under accident conditions, the licensee shall reexamine ventilation intakes, location of potential plant leakage (ingress and egress), and control room filter capabilities, and submit the results of this review to the NRC.

3. Emergency action levels shall be revised to require notification of the NRC for all events in the emergency classes described in NUREG-0610, September 1979.

4. The licensee shall comply with the NRC's "INTERIM POSITION FOR CONTAINMENT PURGE AND VENT VALVE OPERATION PENDING RESOLUTION OF ISOLATION VALVE OPERABILITY", as contained in the October 1979 letter to the licensee.

5. Plant personnel shall be trained or retrained in the following areas within thirty days, or prior to startup if required by the Lessons Learned implementation schedule. Plant personnel shall also be retrained in the following areas within thirty days of the time that there are significant changes to the procedures or requirements applicable to these areas:

   Containment and Degraded Core Sampling
   Degraded Core - Training
   Emergency Power for Pressurizer Heaters and Decay Heat Removal
   Containment Isolation
   Containment Purge/Purge Valve Operation
   Subcooling Meter Operation
   Technical Support Center
   Onsite Operational Support Center
   Near-Site Emergency Operations Center
   Emergency Preparedness Plan
   In-Plant Area Airborne Radioiodine Monitors
   Surveillance Testing of Non-ESF Filtration System

6. The licensee shall perform diesel generator testing in accordance with Regulatory Guide 1.108 with a corresponding change in the allowable outage time stipulated in the Limiting Conditions of Operation as follows:

| Numbers of DG Failures In Prior 100 Tests | Test Interval (Days) (R.G. 1.108) | Allowable Outage Time |
|---|---|---|
| 0 or 1 | 30 | As Is |
| 2 | 14 | As Is |
| 3 | 7 | As Is |
| 4 | 3 | 32 hr. |
| 5 | 3 | 8 hr. |
| 6 or more | 3 | None* |

*Plant must achieve hot shutdown with 12 hours and in cold shutdown within the following 30 hours.

7. Requirements regarding reactor operator qualifications shall be revised to incorporate the following for applications submitted after June 1, 1980:

   a. The following experience shall be required for senior operator applicants:

      Applicants for senior operator licensee shall have 4 years of responsible power plant experience. Responsible power plant experience shall be that obtained as a control room operator (fossil or nuclear), field operator (nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of two years' power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least six months of the nuclear power plant experience shall be at the plant for which the applicant seeks a license.

   b. The hot training programs shall be modified so that the training concentrates on the responsibilities and functions of the operator, rather than the senior operator. All individuals who satisfactorily complete this hot training program will be allowed to apply for an operator license. At least three months' experience as a licensed operator is necessary before applying for a senior operator license.

   c. The three month continuous on-the-job training for hot operator applicants shall be as an extra person on shift in the control room. The hot senior operator applicants will have three months continuous on-the-job training as an extra person on shift in training.

   d. In addition to the presently approved training programs, all replacement applicants shall participate in simulator training programs.

   e. Phase II, III and IV cold training program instructors and all hot training program instructors that provide instruction in nuclear power plant operations shall hold senior operator licenses and shall successfully complete applicable requalification programs to maintain their instructor status.

f. In addition to the present operator requalification program requirements, all operator licensees shall participate in periodic retraining and recertification on a full scope simulator representative of Indian Point Units 2 or 3. The frequency of training will be on an annual basis.

C. Within 60 days of the date of the Order, the licensee shall:

1. Review the steady state steam generator operating level to determine the optimum steady state level for the purpose of maximizing dryout time with due consideration for overfilling. The results of this study shall be provided to the NRC.

2. Evaluate possible co-impregnation of the charcoal in the plant's air effluent filtration systems with KI and $I_2$ and an amine such as TEDA (triethylene-diamine) to improve the iodine removal capability of these systems. The results of this review shall be submitted to the NRC.

3. Evaluate effects on plant systems stability if power is reduced as much as 50%, treating power as a parameter. (For example, the effects on the feedwater flow automatic control).

4. Submit a schedule to implement the ATWS instrument modification justified in accordance with the Westinghouse analytical results contained in the letter from T. N. Anderson to S. H. Hanauer in NS-TMA-2182 dated December 30, 1979.

5. Examine methods of establishing the highest reliability for the gas turbines and submit the results to the NRC. The licensee specifically shall:

   (1) Provide details of gas turbine controls, modes of operation, and other relevant information;

   (2) Evaluate possible improvements to the starting and running reliability of the gas turbines;

   (3) Evaluate and initiate actions which will ensure that a gas turbine can be brought on line within one hour after loss of off-site power;

   (4) Determine how gas turbine power can be provided to Indian Point Unit 3; and

(5) Evaluate the limitation that Indian Point Unit 2 not be operated if the gas turbines are out-of-service.

6. Establish an on-site group reporting to offsite management. The function of the group shall be to examine plant operating characteristics, NRC bulletins, Licensing Information Service advisories and other appropriate sources which may indicate areas for improving plant safety. Where useful improvements can be achieved, the group shall also develop and present detailed recommendations for revised procedures, equipment modifications or other improvements.

D. The following measures shall be implemented within 90 days of the date of the Order:

1a. The licensee shall establish the on-site emergency preparedness manning levels on each shift as contained in Table 1 attached to this Appendix.

b. Power Authority and Consolidated Edison shall jointly arrange to provide additional personnel as contained in Table 1 available to the plant on call within 60 minutes.

2. The Power Authority and Consolidated Edison shall jointly review and identify the significant differences between Indian Point Unit 2 and Unit 3 and shall evaluate these differences in light of present regulatory standards and requirements. Consolidated Edison shall provide a justification for the design differences or shall recommend design changes.

3. The licensee shall establish a temporary on-site inter-disciplinary review group consisting of, as a minimum, representatives from the NSSS vendor, the architect-engineer and the plant maintenance and operations staffs. This group shall review and concur in all existing plant emergency procedures. This group shall also review and concur in changes to emergency procedures. Emergency changes may be approved in accordance with current licensee requirements, but shall be subsequently submitted for approval by the review group.

E. The following measures shall be completed within 120 days of the date of the Order:

1. The licensee shall examine key plant system vulnerability areas and possible operator dependent areas with the intent of maximizing the reliability in the subject areas. Specifically, the licensee shall:

a. Verify that the sump for ESF recirculation is free of debris and determine if flow test verification was initially performed. If not performed, explore means to verify. Review existing procedures and training on recirculation alignment and RWST refill.

b. Review administrative check and verification procedures for assuring that the two single failure points (manual) valves in AFWS supply line are in the correct position.

c. Impose an administrative order requiring expeditious shutdown whenever an independent train of the auxiliary feedwater system and any one of the following are inoperable: All backup sources of offsite power, the diesel generator supplying power to the other independent train or either of the other trains of the auxiliary feedwater system.

d. Develop station blackout procedures addressing:

   i. grid dispatcher actions
   ii. reactor operator actions
   iii. diesel generator repairs

e. Assure that DC-powered lighting is available at the steam-turbine driven auxiliary feedwater pump.

f. Verify that the gas turbine station has black-start capability.

g. Review causes for, and procedures and operator training required to diminish, the overall number of reactor and main feedwater trips.

h. Develop or review procedures to restore main feedwater promptly after a trip and to mitigate the consequences of an ATWS event (e.g. emergency boration and CVCS control).

i. Review administrative controls on the manual valve(s) whose misalignment could fail all ECCS.

2. A review of control room emergency procedures shall be conducted for the purpose of improving these procedures from a human factors engineering standpoint. Improvements which can be attained by modifying procedures shall be implemented within the 120 days. Control room displays shall also be reviewed for the purpose of identifying improvements which will increase the operators' ability to assess plant conditions. A report will be submitted to describe the improvements recommended and the schedule for their implementation.

F. Within six months of date of the Order, the licensee shall:

1. Conduct a review of past Licensee Event Reports (LERs) at Indian Point Units 2 and 3. These LERs shall be reviewed to identify design inadequacies (common mode failures, systems interactions, etc.), procedural and training inadequacies, and man-machine/human factor inadequacies. Recommendations shall be submitted for correction of the base cause of the subject LERs. Immediate corrections of deficiencies will be made when possible, with the required notifications to be made to the NRC.

2. Meet meteorological acceptance criteria for emergency preparedness contained in Annex 1 to this Appendix.

3. Conduct a study to determine and document the method by which its plant complies with current safety rules and regulations, in particular those contained in 10 CFR Parts 20 and 50.

4. Evaluate the reliability and failure modes of selected systems/components as follows:

   a. Failure Mode Effects Analysis: Examine the failure modes (random failures and consequences of outages in support systems) of the active components on the reactor coolant pressure boundary. Assess the acceptability of these failure modes.

   b. Implement Failure Mode Effects Analysis for minor departures from operating, maintenance and emergency procedures.

   c. Explore ways to improve the reliability of those components with a particularly high failure rate as delineated in NUREG/CR-1205.

5. Attain full compliance with NRC letters concerning AFWS reliability improvements.

Table 1

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

| Major Functional Area | Major Tasks | Position Title or Expertise | On Shift | Additions Within 60 minutes |
|---|---|---|---|---|
| Plant Operations and Assessment of Operational Aspects | | Shift Supervisor | 1 | -- |
| | | Senior Reactor Operator | 1 | -- |
| | | Control Room Operators 1/ | 2 | -- |
| | | Auxiliary Operators | 2 | |
| Emergency Direction and Control*** (Emergency Coordinator) | | Designated Sr. Official Shift Supervisor or designated facility manager | 1 | -- |
| Notification/ Communication | Notify licensee, State local and Federal personnel & maintain communication | | 1 | 3 |
| Radiological Accident Assessment and Support of Operational Accident Assessment | Emergency Operations Center (EOC) Director | Senior Manager | -- | 1 |
| | EOC Offsite Dose Assessment | Senior Health Physics (HP) Expertise | | 1 |
| | Offsite Surveys | | -- | 4 |
| | Onsite (out-of-plant) | | -- | 2 |
| | In-plant surveys | HP Technicians | 1 | 2 |
| | Chemistry/Radio-chemistry | Rad/Chem Technicians | 1 | 1 |
| Plant System Engineering, Repair and Corrective Actions | Technical Support | Shift Technical Advisor 2/ | 1 | -- |
| | | Core | -- | 1 |
| | | Electrical | -- | 1 |
| | | Mechanical | -- | 1 |
| | Repair and Corrective Actions | Mechanical Maintenance/ | 1** | 1 |
| | | Rad Waste Operator | | 1 |
| | | Electrical Maintenance/ | 1** | 2 |
| | | Instrument and Control (I&C) Technician | | 1 |

Table 1 (contd)

| Major Functional Area | Major Tasks | Position Title or Expertise | On Shift* | Additions Within 60 Minutes |
|---|---|---|---|---|
| Protective Actions (In-Plant) | Radiation Protection:<br>a. Access Control<br>b. HP Coverage for repair, corrective actions, search and rescue first-aid & firefighting<br>c. Personnel monitoring<br>d. Dosimetry | HP Technicians | 2** | 4 |
| Firefighting | -- | -- | Fire Brigade per Technical Specifications | Local Support |
| Rescue Operations and First-Aid | -- | -- | 2** | Local Support |
| Site Access Control and Personnel Accountability | Security, firefighting communications, personnel accountability | Security Personnel | All per Security plan | |
| | | Total | 10 | 26 |

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator. This means that a single unit will require a minimum shift complement of 10, a two-unit complex 13, and a three-unit complex 16.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOC director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

1/ One of the control room operators may be provided by the other Indian Point unit.
2/ For a multi-unit site this function may be filled by a Shift Supervisor or Foreman, provided all other qualification requirements are met.

# METEOROLOGICAL CRITERIA FOR EMERGENCY PREPAREDNESS AT OPERATING NUCLEAR POWER PLANTS

1. Primary Meteorological Measurements Program

   a. Position: All sites with operating nuclear power plants shall have an adequate operational meteorological measurements program to produce real-time and record historical local meteorological data.

   b. Purpose: To allow a determination of the dispersion of radioactive material due to accidental and routine radioactive releases to the atmosphere by the plant.

   c. Acceptance Criteria:

      (1) The meteorological measurements program shall include measurements and calculations of the following parameters:

         (a) Wind direction and speed at a minimum of two levels (see Regulatory Guide 1.23) one of which is representative of the 10-meter level;

         (b) Standard deviation of wind direction fluctuations (sigma theta) at all measured levels;

         (c) Vertical temperature difference for at least one layer;

         (d) Ambient temperature (10 meters);

         (e) Dew point temperature (10 meters);

         (f) Precipitation near ground level; and

         (g) Pasquill stability class used for diffusion estimates.

(2) The remaining acceptance criteria stated in Revision 1, Section 2.3.3 of NUREG-75/087, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, apply.

(3) A quality assurance program shall be established consistent with the applicable provisions of Appendix B to 10 CFR Part 50. The acceptance criteria stated in Revision 1, Section 17.2 of NUREG-75/087 apply.

(4) The meteorological measurements system and associated controlled environment housing for the equipment shall be connected to a power system which is supplied from redundant power sources.

2. Backup Meteorological Measurements Program

a. Position: All sites with operating nuclear power plants shall have a viable backup system and/or procedures to obtain real-time local meteorological data.

b. Purpose: To provide meteorological information when the primary system is out of service, thus providing assurance that basic meteorological information is available during and immediately following an accidental airborne radioactivity release.

c. Acceptance Criteria:

(1) An independent system and/or procedures shall be established for obtaining measurements of wind direction and speed representative of the 10-meter level and a seven category (A-G) estimator of atmospheric stability ($\Delta T$, wind fluctuations, etc.).

NOTE: An independent system is defined as a system installed and maintained by the licensee specifically for the purpose of providing redundant site-specific meteorological information. An independent procedure is defined as a procedure whereby meteorological information can be obtained from an existing well-maintained meteorological installation capable of providing information representative of the site environs.

(2) The systems and/or procedures shall provide information representative of the site environs, and should include data from multiple locations when necessary.

(3) The system and/or procedure shall provide information in a real-time mode in the event necessary parameters from the primary system are not available. Changeover from the primary system to the backup system shall occur within five minutes. This information should be presented in place of the lost record as outlined in Enclosure 1.

(4) The remaining acceptance criteria stated in Revision 1, Section 2.3.3, of NUREG-75/087, apply.

(5) A quality assurance program shall be established consistent with the applicable provisions of Appendix B to 10 CFR Part 50. The acceptance criteria stated in Revision 1, Section 17.2 of NUREG-75/087, apply.

(6) The meteorological measurements and associated controlled environmental housing system for the equipment shall be connected to a power system which is supplied from redundant power sources.

3. Real-time Predictions of Atmospheric Effluent Transport and Diffusion

a. Position: All licensees with operating nuclear power plants shall have a demonstrated system for making real-time, site specific, estimates and predictions of atmospheric effluent transport and diffusion during and immediately following an accidental airborne radioactivity release from the nuclear power plant.

b. Purpose: To provide an input to the assessment of the consequences of accidental radioactive releases to the atmosphere. To aid in the implementation of emergency preparedness decisions.

c. Acceptance Criteria:

(1) Real-time, site specific atmospheric transport and diffusion models shall be developed and used when accidental airborne radioactive releases occur. Two classes of models should be developed; Class A - a model and calculational capability which can produce initial transport and diffusion estimates within fifteen minutes following classification of an incident, and Class B - a model and calculational capability which can produce refined estimates for the duration of the release. The models shall incorporate the following features:

(a) Site area topography, local meteorological anomalies (as at coastal locations) and available local meteorological measurements;

(b) Variations in time and space of the parameters affecting transport and diffusion, including forecasts of changing meteorological conditions, for model Class B only;

(c) Information from all local meteorological measuring systems used in making the transport and diffusion estimates shall be identified. The licensee shall make arrangements to transmit data from these systems at 30-minute intervals duing an incident.

(2) The transport and diffusion estimates shall include current and forecast plume position, dimensions and radioactivity concentrations at 30-minute intervals as a minimum. Forecast capability up to 24 hours in the future is required in three-hour increments. Such estimates shall be included as a portion of the information accessible for remote interrogation.

(3) A determination shall be made of the accuracy and conservatism of the models in estimating atmospheric transport and diffusion to distances out to 80 km (50 miles).

4. Remote Interrogation of the Atmospheric Measurement and Prediction Systems

a. Position: All systems producing meteorological data and effluent transport and diffusion estimates at sites with operating nuclear power plants shall have the capability of being remotely interrogated.

b. Purpose: To provide simultaneous real-time meteorological data and transport and diffusion estimates in the site vicinity to the licensee, emergency response organizations and the NRC staff, on demand, during emergency situations.

c. Acceptance Criteria:

(1) The meteorological system shall have the capability of being remotely interrogated simultaneously by the licensee, emergency reponse organization and the NRC.

(2) The meteorological data and effluent transport and diffusion estimates shall be in the format indicated in Enclosure 1.

(3) The systems shall have a dial-up connection for a 300 BAUD ASCII terminal of 80 columns via telephone lines (e.g., output format of RS232C in FSK) and a functional back-up communications link (e.g., radio or satellite).

(4) The system shall have the capability of recalling 15-minute averages of meteorological parameters from at least the previous 12-hour period.

(5) The ~~resolution of the data shall meet the~~ system specifications of accuracy given in Section C.4 of Regulatory Guide 1.23.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of )
)
POWER AUTHORITY OF THE STATE ) Docket No. 50-286
OF NEW YORK )
(Indian Point Station, Unit No. 3) )

## CONFIRMATORY ORDER

I.

The Power Authority of the State of New York (the Licensee) is the holder of Operating License No. DPR-64 (the license) which authorizes operation of the Indian Point Unit No. 3, located in Westchester County, New York, at steady state reactor core power levels not in excess of 3025 megawatts thermal (rated power).

II.

Due to the relatively high population density surrounding the Indian Point site as compared to other nuclear power plant sites, the Indian Point site is believed to present a disproportionately high contribution to the total societal risk from reactor accidents. The NRC Staff (the Staff) has currently under way two separate efforts to address the potential problems posed by this relatively high population density. One of the efforts involves the development, revision, and review of emergency plans. This effort is scheduled to be completed by January 1, 1981.

The other effort is a review of the Indian Point facilities to determine what additional procedural measures and/or design changes can and should be implemented that will further reduce the probability of a severe reactor accident and/or to reduce the consequences of such an accident. Since design changes that may be decided upon will take one to two years to completely install, the Staff has identified a number of extraordinary interim measures that should be accomplished both by the licensees and the Staff. These measures will significantly increase the level of safety at the Indian Point Station and thereby further reduce the probability of a severe reactor accident.

Included among these actions are matters dealing with modes of operations, shift manning levels, enhanced training of operators, and special containment and low pressure interface tests designed to add to the level of safety of operation of the facility. All requirements shall be implemented at the time intervals specified in this Order.

The Licensee, in a letter dated February 1, 1980, has agreed to undertake the actions listed in Appendix A to this Order. It is desirable to confirm the Licensee's commitment by Order.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT:

The licensee perform the actions stated in Appendix A to this Order. The aforementioned actions shall be performed in accordance with the schedule set forth in Appendix A or, in the alternative, the licensee shall place and maintain its facili'y in a cold shutdown condition within 48 hours pending completion of those actions.

IV.

Any person who has an interest affected by this Order may request a hearing within twenty (20) days of the date of the Order. Any request for a hearing will not stay the effectiveness of this Order. Any request for a hearing shall be addressed to Harold R. Denton, Director, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555. If a hearing is requested requested by a person who has an interest affected by this Order, the Commission will issue an Order designating the time and place of any such hearing.

In light of the licensee's expressed willingness to undertake the actions ordered, if a hearing is held concerning this Order, the issue to be considered at the hearing shall be:

> Whether the licensee should perform the actions in Appendix A to this Order in accordance with the schedule stated therein.

Operation of the facility on terms consistent with this Order is not stayed by the pendency of any proceedings on the Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: February 11, 1980
Bethesda, Maryland

APPENDIX A

A. The licensee shall:

1. Maintain reactor power level as necessary such that calculated fuel peak clad temperature does not exceed 2000°F under large break LOCA conditions.

2. Revise plant operating procedures as necessary to require a base load mode type of operation only, without load following.

3. Conduct a low presssure gross leak test of containment prior to any start up from cold shutdown conditions. If other means can be found to verify containment integrity, the licensee may propose such procedures to the Commission for its review and approval.

4. Maintain at least two senior reactor operators (SROs), one of whom may be the shift supervisor, in the control room at all times during power operations or hot shutdown, except that the shift supervisor shall be allowed to leave the immediate vicinity of the control room as duties may require, provided he is available to respond to an emergency by returning to the control room within ten minutes. The shift or watch supervisor's office is considered part of the control room.

5. Conduct testing to assure that the LPI/RHR check valves are in fact installed correctly and functioning as pressure isolation barriers when the plant is at pressure and producing power. Verification of valve operability shall be performed prior to plant restart if shutdown at the time of issuance of the Order and thereafter whenever RCS pressure has decreased to within 100 psig of RHR system design pressure.

6. Submit not later than March 1, 1980 the results of a review of possible permanent plant modifications and procedures to further reduce the potential of a severe reactor accident and resultant radiation releases.

7. Require that all reactor operators and senior reactor operators conduct simulator training and in-plant walk through of the following emergency procedures. The in-plant walk-throughs shall be completed prior to the next reactor startup following issuance of the Order, or within thirty days of the date of issuance, whichever occurs first. Those reactor operators and senior reactor operators who have not received simulator training within the past three months on these items shall be given such simulator training within 60 days of the date of the Order:

a. Plant or reactor startups to include a range wherein reactivity feedback from nuclear heat addition is noticeable and heat up rate is established

b. Manual control of steam generator level and/or feedwater during startup and shutdown

c. Any significant (10%) power change using manual rod control

d. Loss of Coolant

(i) including significant PWR steam generator leaks

(ii) inside and outside containment

(iii) large and small, including leak rate determination

(iv) saturated reactor coolant response (PWR)

e. Loss of core coolant flow/natural circulation

f. Loss of all feedwater (normal and emergency)

g. Station blackout

h. Anticipated Transients Without Scram (ATWS)

i. Stuck open relief valve on secondary side

j. Intersystem LOCA

B. The licensee shall implement the following measures within 30 days of the date of the Order:

1. A vendor representative will be stationed on site for engineering consultation at Indian Point Unit 2 and Unit 3 on plant operations and maintenance to increase plant safety. The representative shall be from the NSSS vendor, architect/engineering or start up engineering firm.

2. To ensure control room habitability under accident conditions, the licensee shall reexamine ventilation intakes, location of potential plant leakage (ingress and egress), and control room filter capabilities, and submit the results of this review to the NRC.

3. Emergency action levels shall be revised to require notification of the NRC for all events in the emergency classes described in NUREG-0610, September 1979.

4. The licensee shall comply with the NRC's "INTERIM POSITION FOR CONTAINMENT PURGE AND VENT VALVE OPERATION PENDING RESOLUTION OF ISOLATION VALVE OPERABILITY", as contained in the October 1979 letter to the licensee.

5. Plant personnel shall be trained or retrained in the following areas within thirty days, or prior to startup if required by the Lessons Learned implementation schedule. Plant personnel shall also be retrained in the following areas within thirty days of the time that there are significant changes to the procedures or requirements applicable to these areas:

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   Emergency Power for Pressurizer Heaters and Decay Heat Removal
   Containment Isolation
   Containment Purge/Purge Valve Operation
   Subcooling Meter Operation
   Technical Support Center
   Onsite Operational Support Center
   Near-Site Emergency Operations Center
   Emergency Preparedness Plan
   In-Plant Area Airborne Radioiodine Monitors
   Surveillance Testing of Non-ESF Filtration System

6. The licensee shall perform diesel generator testing in accordance with Regulatory Guide 1.108 with a corresponding change in the allowable outage time stipulated in the Limiting Conditions of Operation as follows:

| Numbers of DG Failures In Prior 100 Tests | Test Interval (Days) (R.G. 1.108) | Allowable Outage Time |
|---|---|---|
| 0 or 1 | 30 | As Is |
| 2 | 14 | As Is |
| 3 | 7 | As Is |
| 4 | 3 | 32 hr. |
| 5 | 3 | 8 hr. |
| 6 or more | 3 | None* |

*Plant must achieve hot shutdown with 12 hours and in cold shutdown within the following 30 hours.

7. Requirements regarding reactor operator qualifications shall be revised to incorporate the following for applications submitted after June 1, 1980:

   a. The following experience shall be required for senior operator applicants:

      Applicants for senior operator licensee shall have 4 years of responsible power plant experience. Responsible power plant experience shall be that obtained as a control room operator (fossil or nuclear), field operator (nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of two years' power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least six months of the nuclear power plant experience shall be at the plant for which the applicant seeks a license.

   b. The hot training programs shall be modified so that the training concentrates on the responsibilities and functions of the operator, rather than the senior operator. All individuals who satisfactorily complete this hot training program will be allowed to apply for an operator license. At least three months' experience as a licensed operator is necessary before applying for a senior operator license.

   c. The three month continuous on-the-job training for hot operator applicants shall be as an extra person on shift in the control room. The hot senior operator applicants will have three months continuous on-the-job training as an extra person on shift in training.

   d. In addition to the presently approved training programs, all replacement applicants shall participate in simulator training programs.

   e. Phase II, III and IV cold training program instructors and all hot training program instructors that provide instruction in nuclear power plant operations shall hold senior operator licenses and shall successfully complete applicable requalification programs to maintain their instructor status.

f. In addition to the present operator requalification program requirements, all operator licensees shall participate in periodic retraining and recertification on a full scope simulator representative of Indian Point Units 2 or 3. The frequency of training will be on an annual basis.

C. Within 60 days of the date of the Order, the licensee shall:

1. Review the steady state steam generator operating level to determine the optimum steady state level for the purpose of maximizing dryout time with due consideration for overfilling. The results of this study shall be provided to the NRC.

2. Evaluate possible co-impregnation of the charcoal in the plant's air effluent filtration systems with KI and $I_2$ and an amine such as TEDA (triethylene-diamine) to improve the iodine removal capability of these systems. The results of this review shall be submitted to the NRC.

3. Evaluate effects on plant systems stability if power is reduced as much as 50%, treating power as a parameter. (For example, the effects on the feedwater flow automatic control).

4. Submit a schedule to implement the ATWS instrument modification justified in accordance with the Westinghouse analytical results contained in the letter from T. N. Anderson to S. H. Hanauer in NS-TMA-2182 dated December 30, 1979.

5. Examine methods of establishing the highest reliability for the gas turbines and submit the results to the NRC. The licensee specifically shall:

   (1) Provide details of gas turbine controls, modes of operation, and other relevant information;

   (2) Evaluate possible improvements to the starting and running reliability of the gas turbines;

   (3) Evaluate and initiate actions which will ensure that a gas turbine can be brought on line within one hour after loss of off-site power;

   (4) Determine how gas turbine power can be provided to Indian Point Unit 3; and

(5) Evaluate the limitation that Indian Point Unit 2 not be operated if the gas turbines are out-of-service.

6. Establish an on-site group reporting to offsite management. The function of the group shall be to examine plant operating characteristics, NRC bulletins, Licensing Information Service advisories and other appropriate sources which may indicate areas for improving plant safety. Where useful improvements can be achieved, the group shall also develop and present detailed recommendations for revised procedures, equipment modifications or other improvements.

D. The following measures shall be implemented within 90 days of the date of the Order:

1a. The licensee shall establish the on-site emergency preparedness manning levels on each shift as contained in Table 1 attached to this Appendix.

b. Power Authority and Consolidated Edison shall jointly arrange to provide additional personnel as contained in Table 1 available to the plant on call within 60 minutes.

2. The Power Authority and Consolidated Edison shall jointly review and identify the significant differences between Indian Point Unit 2 and Unit 3 and shall evaluate these differences in light of present regulatory standards and requirements. Consolidated Edison shall provide a justification for the design differences or shall recommend design changes.

3. The licensee shall establish a temporary on-site inter-disciplinary review group consisting of, as a minimum, representatives from the NSSS vendor, the architect-engineer and the plant maintenance and operations staffs. This group shall review and concur in all existing plant emergency procedures. This group shall also review and concur in changes to emergency procedures. Emergency changes may be approved in accordance with current licensee requirements, but shall be subsequently submitted for approval by the review group.

E. The following measures shall be completed within 120 days of the date of the Order:

1. The licensee shall examine key plant system vulnerability areas and possible operator dependent areas with the intent of maximizing the reliability in the subject areas. Specifically, the licensee shall:

a. Verify that the sump for ESF recirculation is free of debris and determine if flow test verification was initially performed. If not performed, explore means to verify. Review existing procedures and training on recirculation alignment and RWST refill.

b. Review administrative check and verification procedures for assuring that the two single failure points (manual) valves in AFWS supply line are in the correct position.

c. Impose an administrative order requiring expeditious shutdown whenever an independent train of the auxiliary feedwater system and any one of the following are inoperable: All backup sources of offsite power, the diesel generator supplying power to the other independent train or either of the other trains of the auxiliary feedwater system.

d. Develop station blackout procedures addressing:

   i. grid dispatcher actions
   ii. reactor operator actions
   iii. diesel generator repairs

e. Assure that DC-powered lighting is available at the steam-turbine driven auxiliary feedwater pump.

f. Verify that the gas turbine station has black-start capability.

g. Review causes for, and procedures and operator training required to diminish, the overall number of reactor and main feedwater trips.

h. Develop or review procedures to restore main feedwater promptly after a trip and to mitigate the consequences of an ATWS event (e.g. emergency boration and CVCS control).

i. Review administrative controls on the manual valve(s) whose misalignment could fail all ECCS.

2. A review of control room emergency procedures shall be conducted for the purpose of improving these procedures from a human factors engineering standpoint. Improvements which can be attained by modifying procedures shall be implemented within the 120 days. Control room displays shall also be reviewed for the purpose of identifying improvements which will increase the operators' ability to assess plant conditions. A report will be submitted to describe the improvements recommended and the schedule for their implementation.

F. Within six months of date of the Order, the licensee shall:

1. Conduct a review of past Licensee Event Reports (LERs) at Indian Point Units 2 and 3. These LERs shall be reviewed to identify design inadequacies (common mode failures, systems interactions, etc.), procedural and training inadequacies, and man-machine/human factor inadequacies. Recommendations shall be submitted for correction of the base cause of the subject LERs. Immediate corrections of deficiencies will be made when possible, with the required notifications to be made to the NRC.

2. Meet meteorological acceptance criteria for emergency preparedness contained in Annex 1 to this Appendix.

3. Conduct a study to determine and document the method by which its plant complies with current safety rules and regulations, in particular those contained in 10 CFR Parts 20 and 50.

4. Evaluate the reliability and failure modes of selected systems/components as follows:

   a. Failure Mode Effects Analysis: Examine the failure modes (random failures and consequences of outages in support systems) of the active components on the reactor coolant pressure boundary. Assess the acceptability of these failure modes.

   b. Implement Failure Mode Effects Analysis for minor departures from operating, maintenance and emergency procedures.

   c. Explore ways to improve the reliability of those components with a particularly high failure rate as delineated in NUREG/CR-1205.

5. Attain full compliance with NRC letters concerning AFWS reliability improvements.

Table 1

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

| Major Functional Area | Major Tasks | Position Title or Expertise | On Shift | Additions Within 60 minutes |
|---|---|---|---|---|
| Plant Operations and Assessment of Operational Aspects | | Shift Supervisor | 1 | -- |
| | | Senior Reactor Operator | 1 | -- |
| | | Control Room Operators 1/ | 2 | -- |
| | | Auxiliary Operators | 2 | |
| Emergency Direction and Control*** (Emergency Coordinator) | | Designated Sr. Official Shift Supervisor or designated facility manager | 1 | -- |
| Notification/ Communication | Notify licensee, State local and Federal personnel & maintain communication | | 1 | 3 |
| Radiological Accident Assessment and Support of Operational Accident Assessment | Emergency Operations Center (EOC) Director | Senior Manager | -- | 1 |
| | EOC Offsite Dose Assessment | Senior Health Physics (HP) Expertise | | 1 |
| | Offsite Surveys | | -- | 4 |
| | Onsite (out-of-plant) | | -- | 2 |
| | In-plant surveys | HP Technicians | 1 | 2 |
| | Chemistry/Radio-chemistry | Rad/Chem Technicians | 1 | 1 |
| Plant System Engineering, Repair and Corrective Actions | Technical Support | Shift Technical Advisor 2/ | 1 | -- |
| | | Core | -- | 1 |
| | | Electrical | -- | 1 |
| | | Mechanical | -- | 1 |
| | Repair and Corrective Actions | Mechanical Maintenance/ | 1** | 1 |
| | | Rad Waste Operator | | 1 |
| | | Electrical Maintenance/ | 1** | 2 |
| | | Instrument and Control (I&C) Technician | | 1 |

Table 1 (contd)

| Major Functional Area | Major Tasks | Position Title or Expertise | On Shift* | Additions Within 60 Minutes |
|---|---|---|---|---|
| Protective Actions (In-Plant) | Radiation Protection:<br>a. Access Control<br>b. HP Coverage for repair, corrective actions, search and rescue first-aid & firefighting<br>c. Personnel monitoring<br>d. Dosimetry | HP Technicians | 2** | 4 |
| Firefighting | -- | -- | Fire Brigade per Technical Specifications | Local Support |
| Rescue Operations and First-Aid | -- | -- | 2** | Local Support |
| Site Access Control and Personnel Accountability | Security, firefighting communications, personnel accountability | Security Personnel | All per Security plan | |
| | | Total | 10 | 26 |

Notes:

* For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator. This means that a single unit will require a minimum shift complement of 10, a two-unit complex 13, and a three-unit complex 16.

** May be provided by shift personnel assigned other functions.

*** Overall direction of facility response to be assumed by EOC director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.

1/ One of the control room operators may be provided by the other Indian Point unit.

2/ For a multi-unit site this function may be filled by a Shift Supervisor or Foreman, provided all other qualificatión requirements are met.

# METEOROLOGICAL CRITERIA FOR EMERGENCY PREPAREDNESS AT OPERATING NUCLEAR POWER PLANTS

1. Primary Meteorological Measurements Program

   a. Position: All sites with operating nuclear power plants shall have an adequate operational meteorological measurements program to produce real-time and record historical local meteorological data.

   b. Purpose: To allow a determination of the dispersion of radioactive material due to accidental and routine radioactive releases to the atmosphere by the plant.

   c. Acceptance Criteria:

      (1) The meteorological measurements program shall include measurements and calculations of the following parameters:

         (a) Wind direction and speed at a minimum of two levels (see Regulatory Guide 1.23) one of which is representative of the 10-meter level;

         (b) Standard deviation of wind direction fluctuations (sigma theta) at all measured levels;

         (c) Vertical temperature difference for at least one layer;

         (d) Ambient temperature (10 meters);

         (e) Dew point temperature (10 meters);

         (f) Precipitation near ground level; and

         (g) Pasquill stability class used for diffusion estimates.

(2) The remaining acceptance criteria stated in Revision 1, Section 2.3.3 of NUREG-75/087, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, apply.

(3) A quality assurance program shall be established consistent with the applicable provisions of Appendix B to 10 CFR Part 50. The acceptance criteria stated in Revision 1, Section 17.2 of NUREG-75/087 apply.

(4) The meteorological measurements system and associated controlled environment housing for the equipment shall be connected to a power system which is supplied from redundant power sources.

2. Backup Meteorological Measurements Program

a. Position: All sites with operating nuclear power plants shall have a viable backup system and/or procedures to obtain real-time local meteorological data.

b. Purpose: To provide meteorological information when the primary system is out of service, thus providing assurance that basic meteorological information is available during and immediately following an accidental airborne radioactivity release.

c. Acceptance Criteria:

(1) An independent system and/or procedures shall be established for obtaining measurements of wind direction and speed representative of the 10-meter level and a seven category (A-G) estimator of atmospheric stability ($\Delta T$, wind fluctuations, etc.).

NOTE: An independent system is defined as a system installed and maintained by the licensee specifically for the purpose of providing redundant site-specific meteorological information. An independent procedure is defined as a procedure whereby meteorological information can be obtained from an existing well-maintained meteorological installation capable of providing information representative of the site environs.

(2) The systems and/or procedures shall provide information representative of the site environs, and should include data from multiple locations when necessary.

(3) The system and/or procedure shall provide information in a real-time mode in the event necessary parameters from the primary system are not available. Changeover from the primary system to the backup system shall occur within five minutes. This information should be presented in place of the lost record as outlined in Enclosure 1.

(4) The remaining acceptance criteria stated in Revision 1, Section 2.3.3, of NUREG-75/087, apply.

(5) A quality assurance program shall be established consistent with the applicable provisions of Appendix B to 10 CFR Part 50. The acceptance criteria stated in Revision 1, Section 17.2 of NUREG-75/087, apply.

(6) The meteorological measurements and associated controlled environmental housing system for the equipment shall be connected to a power system which is supplied from redundant power sources.

3. Real-time Predic ons of Atmospheric Effluent Transport and Diffusion

a. Position: All licensees with operating nuclear power plants shall have a demonstrated system for making real-time, site specific, estimates and predictions of atmospheric effluent transport and diffusion during and immediately following an accidental airborne radioactivity release from the nuclear power plant.

b. Purpose: To provide an input to the assessment of the consequences of accidental radioactive releases to the atmosphere. To aid in the implementation of emergency preparedness decisions.

c. Acceptance Criteria:

(1) Real-time, site specific atmospheric transport and diffusion models shall be developed and used when accidental airborne radioactive releases occur. Two classes of models should be developed; Class A - a model and calculational capability which can produce initial transport and diffusion estimates within fifteen minutes following classification of an incident, and Class B - a model and calculational capability which can produce refined estimates for the duration of the release. The models shall incorporate the following features:

(a) Site area topography, local meteorological anomalies (as at coastal locations) and available local meteorological measurements;

(b) Variations in time and space of the parameters affecting transport and diffusion, including forecasts of changing meteorological conditions, for model Class B only;

(c) Information from all local meteorological measuring systems used in making the transport and diffusion estimates shall be identified. The licensee shall make arrangements to transmit data from these systems at 30-minute intervals duing an incident.

(2) The transport and diffusion estimates shall include current and forecast plume position, dimensions and radioactivity concentrations at 30-minute intervals as a minimum. Forecast capability up to 24 hours in the future is required in three-hour increments. Such estimates shall be included as a portion of the information accessible for remote interrogation.

(3) A determination shall be made of the accuracy and conservatism of the models in estimating atmospheric transport and diffusion to distances out to 80 km (50 miles).

4. Remote Interrogation of the Atmospheric Measurement and Prediction Systems

a. Position: All systems producing meteorological data and effluent transport and diffusion estimates at sites with operating nuclear power plants shall have the capability of being remotely interrogated.

b. Purpose: To provide simultaneous real-time meteorological data and transport and diffusion estimates in the site vicinity to the licensee, emergency response organizations and the NRC staff, on demand, during emergency situations.

c. Acceptance Criteria:

(1) The meteorological system shall have the capability of being remotely interrogated simultaneously by the licensee, emergency reponse organization and the NRC.

(2) The meteorological data and effluent transport and diffusion estimates shall be in the format indicated in Enclosure 1.

(3) The systems shall have a dial-up connection for a 300 BAUD ASCII terminal of 80 columns via telephone lines (e.g., output format of RS232C in FSK) and a functional back-up communications link (e.g., radio or satellite).

(4) The system shall have the capability of recalling 15-minute averages of meteorological parameters from at least the previous 12-hour period.

(5) The resolution of the data shall meet the system specifications of accuracy given in Section C.4 of Regulatory Guide 1.23.

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

December 17, 1979

The Honorable Morris Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D. C. 20515

Dear Mr. Chairman:

I am writing on behalf of the Commission in response to your request of December 11 for our views on Section 104(a)(6). This section was offered as an amendment to H.R. 2608 by Representative Bingham and adopted by the full House. We appreciate this opportunity to provide our comments on this proposal prior to House-Senate conference.

We fully endorse what we believe to be the intent of the Bingham amendment, i.e., that the NRC undertake a comprehensive program for systematically reevaluating the safety of all currently operating plants. However, a majority of the Commissioners (Chairman Ahearne, Commissioner Kennedy and Commissioner Hendrie) does not believe the Bingham amendment would represent the most effective use of resources in accomplishing this goal.

The amendment is intended to provide information concerning the degree to which operating power plants conform to NRC standards and criteria required of current applicants for operating licenses and construction permits as well as to provide a schedule for resolving generic safety issues identified in NUREG 0410.

In developing this information one approach would be to conduct a quick review which would not involve substantial resources. This type of review would probably be based on telephone calls to licensees, comparison of licensing dates with effective dates of NRC requirements, and a review of outstanding issues. This was the approach the NRC staff originally had in mind for complying with the amendment. However, upon further consideration of how this process would actually work, we have concluded this alternative would have limited usefulness in evaluating safety. To be useful, the information developed by the industry and the NRC staff would have to be carefully compiled and reviewed for completeness and accuracy. For example, under the first approach one might assume plants licensed before the effective date of a requirement do not meet that requirement. However, a closer look might show that the NRC imposed the requirement on licensees while it was still a draft position.

Thus, careful examination of each plant would be necessary to accurately determine the status of each requirement. Following this second approach for all operating plants is estimated to require six months and involve approximately 15 man-years of NRC effort. This is a significant resources allocation decision which will have an impact on the ongoing Three Mile Island efforts.

The majority of the Commissioners would prefer a third approach.

As you are aware, two years ago the Commission undertook a reevaluation on a limited basis with respect to 11 of the older operating plants. We believe a variation of this Systematic Evaluation Program should be developed for application to all operating plants. Such a program should also address generic safety issues. A specific task to accomplish this objective has already been included in the proposed Three Mile Island Task Action Plans currently under review by the Commission.

It will take several months for the NRC staff to develop and propose, and for the Commission to approve, this systematic program for evaluating the safety of all operating plants. It most likely will include some elements of the ongoing Systematic Evaluation Program, in which evaluations are being made by the NRC staff of the design of the older plants with regard to some 130 safety "topics," e.g., determining the adequacy of plant design with respect to geologic and seismologic phenomena (earthquakes, land slides, ground collapse, liquefaction, etc.). This would be in contrast to evaluating the safety of these plants by comparing and contrasting their design features with all current staff standards and criteria. The new program will undoubtedly also contain some elements which do involve a comparison of existing plant design features against some of the more safety significant current NRC requirements for the design of these features.

Our preference therefore would be to replace the current Section 104(a)(6) with language along the following lines:

> "The NRC shall develop and provide to the Congress within 120 days a comprehensive plan for the systematic safety evaluation of all currently operating plants. The Commission shall forward to the Congress a report on the progress on implementation of the evaluation program prior to February 1, 1981, as a separate document, and for each succeeding year as a separate chapter of the Commission's annual report (required under Section 307(c) of the Energy Reorganization Act of 1974)."

If this alternative is not acceptable to the conferees, we recommend that the time period for compliance with the current Section 104(a)(6) be extended to 180 days to minimize any interference it will have with ongoing Three Mile Island-related efforts.

In addition to your request for our views, your letter expresses concern that NRC staff doubts about the Bingham amendment were not made explicit to Congress or transmitted in writing to your Subcommittee in a timely manner, presumably before floor action on H.R. 2608.

Several days before Mr. Bingham offered his amendment on the House floor, similar language was circulated to senior staff of the Office of Nuclear Reactor Regulation, who offered the informal opinion that the information required by the amendment could be compiled but that six months rather than four would afford a more reasonable time frame for implementation. This informal opinion was conveyed to Mr. Bingham's staff and to the staff of your Subcommittee.

The NRC officials involved did not volunteer an opinion as to whether such a compilation represented the most effective use of limited resources. At the time the staff interpreted the Bingham amendment as likely to be compatible with the systematic evaluation program to be proposed to the Commission. However, on further consideration, the staff now believes that implementation of the Bingham amendment either would have limited usefulness in evaluating the safety of operating plants or would impact substantially on its ability to develop the reevaluation plan contemplated by the Commission staff.

As previously pointed out, we believe the goal of the amendment is essentially the same as one already set by the Commission. However, it would be difficult to implement the second and third approaches simultaneously because of resource limitations. Thus the current amendment would determine the course of action rather than allow the Commission to reach a decision based on its evaluation of the NRC staff's proposals.

Commissioners Gilinsky and Bradford endorse the Bingham amendment. They regard the requirements of the Bingham amendment as a necessary first step in developing a comprehensive program for the systematic evaluation of currently operating plants. They believe that this information concerning the basic NRC safety requirements to which each operating reactor is subject should be readily available. In their view, the apparent fact that it is not available indicates a surprising disarray in the status of NRC knowledge of operating plants that should not be allowed to continue. They do not interpret the Amendment as requiring any engineering evaluations. Commissioners Gilinsky and Bradford suggest that it would be useful to add to Section 104(a) (6) (B):

> ". . . and which of those items referred to in subparagraph (A) the licensee is required to meet as of the date of this Act. For those cases where a current requirement is not imposed on a licensee, the report should identify the related applicable requirement, if any, and the difference between it and the current requirement;"

This would serve to document what is required of operating plants in light of current standards.

The Commission (Chairman Ahearne and Commissioners Kennedy and Hendrie) has no objection to the Bingham Amendment, if it is understood to be the minimal resource review of the first approach. However, if it is to lead to the significant resource application of the second approach the Commissioners believe the Commission should decide how best to allocate its resources to accomplish the desired goal. They agree with what they perceive to be the intent of the Bingham Amendment but believe their proposed alternative is the proper way to reach a decision on allocation of large staff resources.

Regarding provisions (C) and (E) of the amendment, NUREG 0510 identified which of the issues listed in NUREG 0410 were, in the Commission's judgment, "Unresolved Safety Issues." NUREG 0510 provided estimated programs resolving these issues. As additional items are raised to the level of these issues, schedules are developed for them.

I hope this information is helpful in clarifying the Commission's position.

Sincerely,

John F. Ahearne

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

CHAIRMAN

January 3, 1980

The Honorable Morris K. Udall, Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

The Commission has now had an opportunity to review both versions of S. 562, as passed by the Senate and House. Attached for your convenience and that of the conferees is a table listing major provisions where there are significant differences between the Senate and the House bills and indicating, where appropriate, the NRC preference and the reasons, in abbreviated form, for that preference.

Section 101 of both bills provides spending authority for various NRC program offices for FY 1980. The higher authorizations contained in the House bill reflect the most recent Commission assessment of its needs, taking into account the many necessary changes identified as a result of our evaluation of the Three Mile Island accident. The sum of these items is $426,821,000 for NRC for FY 1980. This amount has the Commission's full support.

The Commission is particularly pleased that both Senate and House included recommended legislation increasing the amount of civil penalties which may be imposed for violations of NRC regulations. We are also pleased that the House version includes a provision (Section 302) long sought by the NRC, which protects certain sensitive, but unclassified, safeguards information, related to the security of nuclear facilities, from public disclosure. The colloquy between Congressman Udall and Congressman Moffett concerning the application of this provision (Congressional Record December 4, 1979 --H-11497-H-11498) has created some uncertainty about the intended coverage of this section. The Commission clearly believes that information on individual shipments (specific times, and, in the few cases where alternative routes are available, the alternative chosen for an individual shipment) should be treated as safeguards information and withheld from public disclosure. We are now considering in an adjudicatory proceeding the issue of whether the routes approved by the NRC should also be kept confidential. The Commission expects to rule on this question in the near future.

In addition to these provisions, several other measures were added which are intended to enhance the NRC's enforcement powers. Section 212 of the Senate bill and 303 of the House bill would provide criminal sanctions for certain acts of nuclear sabotage. Of the two, the Commission prefers the language in the House bill, which is more comprehensive in scope and covers acts of sabotage committed against nuclear fuel in transportation and in storage installations as well as those committed against reactors.

Section 109a of the Senate bill would authorize criminal penalties for knowing violations of NRC safety standards relating to utilization and production facilities. We are concerned that this section could have a chilling effect on individuals who must take action in the event of emergencies or other off-normal situations. In some instances, it might be necessary to violate an NRC safety standard, such as a technical specification or a radiation exposure limit, in order to avoid a more serious occurrence. We would, therefore, prefer more time to consider the possible implications of such a requirement. Commissioner Bradford supports Section 109a provided that emergency situations are adequately recognized.

Section 401 of the House bill makes it a Federal offense to attack construction or quality assurance inspectors at an NRC-licensed project. If the intent of Section 401 is to protect NRC inspectors, we would recommend the language in the attachment. If the intent is to protect licensee/contractor inspectors we believe further study is necessary because Section 1114 of the U.S. Code concerns "Protection of Officers and Employees of the United States."

Section 105 of the House bill and Section 210 of the Senate bill both require the Commission to promulgate rules providing for the notification of State officials of certain types of radioactive waste shipments in or through their states. We would prefer the language in Section 210 of the Senate bill for two reasons. The Senate version allows the Commission to exclude from notification requirements such quantities and types of radioactive wastes as it specifically determines do not pose a potentially significant health and safety hazard. There are approximately 150,000 shipments of radioactive waste each year. Most of these shipments involve small, relatively harmless amounts of material. In addition, we feel that Section 105 of the House bill contains language which would prevent the NRC from protecting specific routing information from disclosure to the general public.

A number of Sections (Sections 108, 202(c), 203, and 210 of the Senate bill) require promulgation of new rules within six months or less. Because of the complexity of the subjects involved and the desirability of permitting public participation in the rulemaking process, it is unlikely the Commission could meet these deadlines. We clearly could not do so without significantly curtailing public participation and the quality of the rule. Consequently, if rulemakings are directed, we recommend that Congress require rules to be proposed, rather than promulgated, within a reasonable time frame--generally six months.

In the case of new siting rules required by Section 108 (Senate) the staff estimates that it could not develop a technically defensible proposed rule in less than nine months. Moreover, several criteria (i.e. fission product release and resultant radiation exposure) set out in Section 108 (as passed by the Senate in July) have proven less useful in measuring site suitability than originally envisioned. We recommend that the bill not prescribe specific criteria for siting regulations, but allow them to be developed in a public rulemaking.

Section 202 of the Senate bill and Section 104 of the House bill address improved emergency planning around nuclear power plants. The conferees should be aware that the Commission has underway a rulemaking related to emergency planning. On December 19, the Federal Register published a Commission notice of a proposed rule for public comment. The proposed rule would require NRC concurrence in State and local emergency response plans as a condition for issuing an operating license. It also contains several alternatives for existing operating plants. One alternative would require the automatic shutdown of operating plants no later than January 1, 1981 unless the NRC has concurred in State or local plans or a specific exemption is granted by the NRC. While this approach is similar to that included in the Senate bill, the Commission would prefer to consider these requirements without statutory language and to be able to make determinations for exemption on a case-by-case basis. Commissioners Gilinsky and Bradford have no objection to Section 202 of the Senate bill.

NRC review teams will be visiting all operating reactor sites within the next seven months to assess the preparedness of utilities and to some degree, State and local governments. Teams have already visited 22 of the sites--generally in the most populated areas. We believe that, using the information obtained during site reviews and working closely with the Federal Emergency Management Agency (FEMA), we would be able to prepare the type of report contemplated in Section 104(a) (3) of the House bill for the operating reactor sites nine months after enactment. The Commission is considering whether some construction permits, which have already been issued, should be reconsidered because of the emergency planning considerations of the proposed rule. If the conferees wish to include construction permits in Section 104(a) (3), then the Commission would prefer that the language be modified to be limited to certain selected construction permits which pose potential difficulties in evacuation, for example, sites in heavily populated areas. The Commission would also request that the reporting time for the construction permit reviews be extended to 12 months from enactment. In all cases, the Commission intends to conduct such a review at each construction site prior to issuing the operating license.

In his December 7 statement on nuclear energy, President Carter announced that Executive Branch responsibility for offsite radiological emergency planning and response would be assumed by FEMA. It is too early to clearly define the respective roles of FEMA and NRC in the reviews called for by Section 104(a) (3) at this time. Accordingly, the conferees may wish to consider the role of FEMA in this regard.

During floor consideration, the House adopted an amendment offered by Mr. Bingham which adds Section 104(a) (6) requiring the Commission to compile certain information relating to the compliance of existing operating plants with current Commission safety standards and regulations. While the Commission does not oppose this amendment, a majority would prefer a different approach, as outlined in the attached letter to Chairman Udall.

Several other provisions (Section 205(c), 206 and 207 of the Senate bill) establish requirements for studies and reports to Congress with deadlines which the Commission does not believe it can meet. Alternative dates are suggested in the attached table.

As a general note, the Commission may have difficulty meeting the various reporting requirements and deadlines. We are taking a comprehensive look at our current methods of regulation. In particular, we are developing an Action Plan which will consolidate and prioritize all of the issues which have been identified as a result of the various TMI reviews. General approval of the Action Plan is anticipated by February 15, 1980. I should note that implementation of this Action Plan will involve a significant effort. Preliminary estimates include several hundred manyears of NRC staff effort. Many of the reporting requirements in the bills involve tasks which are included in the proposed Action Plan. Therefore, the Commission would prefer to address these requirements in the context of the Action Plan as the most effective and efficient use of resources, rather than treating them on an individual basis.

I trust that these comments will be helpful to the conferees in their consideration of S. 562. If the Commission can be of any additional assistance, please feel free to call upon us.

Sincerely,

J F Ahearne

John F. Ahearne

Enclosures:
As stated

cc: Congressman Harley O. Staggers
Congressman Jonathan B. Bingham
Congressman John Dingell
Congressman Philip R. Sharp
Congressman Bruce Vento
Congressman James Weaver
Congressman Edward Markey
Congressman Steven Symms
Congressman Manuel Lujan
Congressman Clarence J. Brown
Congressman Tom Corcoran

Senator Gary Hart
Senator Alan Simpson
Senator Jennings Randolph
Senator Pete Domenici
Senator Daniel Moynihan

# NRC's TABLE OF COMPARSIONS

| MAJOR PROVISIONS | SENATE BILL | HOUSE BILL | NRC PREFERENCE |
|---|---|---|---|
| Funding Authorized | $398,300,000 (Sec. 101) | $426,821,000 (Sec. 101) | $426,821,000. This amount reflects the most recent NRC assessment of its needs taking into account the impact of TMI. |
| Siting | NRC to promulgate new siting rule based on certain specified criteria in 6 mos. (Sec. 108) | None | If rule required, would prefer 9 mos. to publish proposed rule and that no specific criteria be prescribed in the statute. NRC staff estimates that it will take 9 mos. to prepare a technically defensible proposed rule. Promulgation within 6 mos. would require a severe curtailment of public participation in the rulemaking. |
| Criminal Sanctions for Knowing Violation of NRC Safety Standard | Makes it a crime to knowingly violate an NRC safety standard relating to utilization or production facilities (Sec. 109) | None | NRC would prefer that this be deferred until further study. Preliminary study by NRC staff raise several concerns. It is believed the section as drafted may have a chilling effect on individuals and prevent action which may be proper under emergency circumstances. It may in some instances be necessary to violate a safety standard and risk one set of consequences in order to avoid a more serious set of consequences. |
| State Emergency Response Plans | Makes issuance of operating licenses contingent on NRC concurrence in State emergency plan. Existing plants would have to cease operation by 6/1/80 if State of situs does not have a concurred in plan. Requires | Requires NRC to set standards for State plans by rule, review State plans of affected States, assess State's ability to carry them out, notify governors if plans do not conform to NRC guidelines and report to Congress on the results. (Sec. 104a(1)-(5)). | NRC currently has underway a rulemaking on emergency planning. One alternative in the proposed rule published 12/19/79 would require the automatic shutdown of any operating plant in a State without an NRC concurred |

| MAJOR PROVISIONS | SENATE BILL | HOUSE BILL | NRC PREFERENCE |
| --- | --- | --- | --- |
| State Emergency Response Plan (continued) | NRC to promulgate minimum requirements for State plans within 6 mos. of enactment. (Sec. 202) | | In plan by no later than 1/1/81 unless NRC grants a specific exemption. NRC prefers the flexibility to make rule without statutory language. |
| | | A related provision (Sec. 104a(3)) requires NRC to assess adequacy of emergency planning for each site & report to Congress within 6 mos. | NRC believes 9 mos. needed to complete assessments of operating reactors sites and make a report. The Commission would prefer to review CP's in critical areas after the OL reviews and prepare a report in 12 mos. In all cases, CP's will be reviewed prior to issuing an OL. The roles of FEMA and NRC should be clarified prior to statutory assignments of authority. |
| List of Standards Imposed on Existing Plants | | (Sec. 104a(6)) directs NRC to compile certain information on the degree to which existing operating plants have been required to meet current NRC safety standards and regulations. | A majority of the Commission would prefer to achieve the same objective by completing an NRC plan of action already underway. It recommends consideration of alternative language requiring NRC to provide Congress within 120 days a "comprehensive plan for the systematic safety evaluation of all currently operating plants" with a progress report required by 2/1/81 and in the annual report thereafter. Two Commissioners endorse Sec. 104a(6) but suggest some additional language to document what is required of operating plants in light of current standards. Commission comments are included in a letter to Chairman Udall dated 12/17/79. |
| NRC Emergency Response Plan | NRC to promulgate by rule within 6 mos. a contingency plan detailing NRC emergency response to an extraordinary nuclear occurrence. (Sec. 203) | | If a rule is required, NRC recommends 6 mos. to publish a proposed rule rather than to promulgate a rule. Promulgation within 6 mos. could not be done without severely curtailing public participation. |

| MAJOR PROVISIONS | SENATE BILL | HOUSE BILL | NRC PREFERENCE |
| --- | --- | --- | --- |
| Emergency Monitoring Plan | NRC to send to Congress within 90 days a plan for remote & instantaneous monitoring of principal safety instruments at all plants. (Sec. 205) | None | NRC Staff believes this task cannot be accomplished in the time frame prescribed. Development and implementation of such a plan is part of the proposed NRC action plan. If this requirement is retained, it recommends 6 mos. for submission of plan to Congress. |
| Emergency Communications Study and Report | NRC to study emergency communications in 30-day period following TMI and report to Congress by 1/1/80 with recommendations. (Sec. 206) | None | If this requirement is retained, NRC recommends 6 mos. from date of enactment as reasonable time for reporting to Congress. |
| Plan for Improved Operator Training & Licensing | NRC must submit to Congress within 6 mos. a plan for improving training, retraining & licensing of reactor operators in accord with certain specifications. (Sec. 207a.) | None | Development and implementation of such a plan is part of the proposed NRC action plan. If this requirement is retained, NRC recommends 9 mos. for submitting plan to Congress. |
| Study of Licensing Senior Plant Officials | NRC must study the feasibility & value of licensing plant managers & senior officials & report to Congress with findings & recommendations in 6 mos. (Sec. 207b) | None | If this requirement is retained, NRC staff notes that the study will have to be contracted out because of limited staff resources. It estimates that the contract award, the study, and NRC review cannot be completed in 6 mos. & recommends 12 mos. from enactment for submitting report to Congress. |
| Notice to States of Certain Waste Shipments | NRC must promulgate rules by 10/1/79 whereby notice is given to States of nuclear waste shipments in or through the State. A proviso permits NRC to exempt types & quantities of shipments which it specifically determines not to pose a significant hazard. (Sec. 210) | NRC must promulgate notice to States rules within 90 days. Notification is not "safeguards information" for purposes of new section 147 on safeguards information. (Sec. 105) | The Commission prefers the Senate version with a proviso permitting it to exempt certain types & quantities of waste. It notes that there are approx. 150,000 waste shipments annually. Most of these involve small, relatively harmless quantities & types of material. |

| MAJOR PROVISIONS | SENATE BILL | HOUSE BILL | NRC PREFERENCE |
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| Notice to States of Certain Waste Shipments (cont.) | | | The Commission feels that Sec. 105 of the House Bill would prevent the NRC from withholding from public disclosure, any specific routing information which is provided to Governors. The House language indicates that such notices are not "safeguards information, "but allows the NRC to require that the Governors keep the information confidential. The Commission, therefore, would not be able to withhold the notifications from public disclosure under the provisions of Section 147 of the Act. The Commission believes that specific routes and times of shipments should be treated as safeguards information.<br>If Sec. 105 is retained, NRC recommends deleting the last 2 sentences of 105a & replacing with the following:<br><br>"Provided, however, that such notification requirements shall not apply to nuclear wastes in such quantities and of such types as the Commission specifically determines do not pose a potentially significant hazard to the health and safety of the public. The Commission may require each Governor receiving such notification comply with the procedures and the standards of confidentiality respecting such notification as the Commission deems necessary pursuant to Section 147 of the Atomic Energy Act of 1954, as amended." |

| MAJOR PROVISIONS | SENATE BILL | HOUSE BILL | NRC PREFERENCE |
|---|---|---|---|
| Sabotage of Nuclear Facilities | Provides for FBI investigation & criminal penalties for acts of sabotage against certain nuclear facilities, including power reactors. (Sec. 212) | Provides criminal penalties for acts of sabotage against nuclear fuel during transportation & at storage installations as well as against facilities covered by Senate bill. (Sec. 303) | NRC prefers the language and broader coverage provided in the House bill. |
| Protection of Certain Inspectors | None | Makes attacks on "any construction inspector or quality assurance inspector" at any NRC licensed project a Federal offense by amending Section 1114 of title 18 of the U.S. Code. (Sec. 401) | NRC is unsure of the exact intent of this provision and would prefer that consideration be postponed until further study.<br><br>As worded, Federal protection appears to be extended to a large category of non-Federal employees. If the intent is to protect licensee & contractor quality assurance personnel, we believe further study is necessary since Section 1114 of the U.S. Code concerns "Protection of officers and employees of the United States".<br><br>If the intent is to protect NRC inspectors, we recommend that 18 U.S.C. 1114 be amended by inserting "any officer or employee of the U.S. Nuclear Regulatory Commission" after "Department of Justice" rather than the present language of Section 401 because many NRC inspectors would not be covered by the current language. |

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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| CONSOLIDATED EDISON COMPANY OF NEW YORK (Indian Point, Unit Nos. 1 and 2)<br><br>POWER AUTHORITY OF THE STATE OF NEW YORK (Indian Point, Unit No. 3) | ) | Docket Nos. 50-3<br>50-247<br>50-286 |

ISSUANCE OF DIRECTOR'S DECISION UNDER 10 CFR 2.206

By petition dated September 17, 1979, the Union of Concerned Scientists (UCS) requested that the Nuclear Regulatory Commission revoke the license for Indian Point Unit No. 1 and order decommissioning of the plant and suspend operation of Indian Point Unit Nos. 2 and 3 pending resolution of various issues cited in the UCS petition. On October 26, 1979, the Comission referred the UCS petition to the NRC staff for treatment pursuant to 10 CFR 2.206 of the Commission's regulations.

Upon consideration of the UCS petition, various statements filed in support of the petition, and other pertinent information, I have granted in part and denied in part the UCS petition. The reasons for this decision are fully described in a "Director's Decision, Under 10 CFR 2.206," which is available for public inspection in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. 20555 and in the local public document room at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601. A copy of this decision will also be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c).

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
1[illegible] day of February, 1980