bcc: OGC Files - Beth/GT/
Docket
REG Files
DRL - Dr. Morris

May 12, 1971

Anthony Z. Roisman, Esq.
Berlin, Roisman & Kessler
1910 N Street, N.W.
Washington, D. C. 20036

In the Matter of Consolidated Edison Company of New York, Inc.
Indian Point Nuclear Generating Unit No. 2
Docket No. 50-247

Dear Mr. Roisman:

Attached herewith are the responses of the AEC Regulatory Staff to the further series of questions (list H and I submitted by you in the above captioned matter).

Sincerely,

Myron Karman
Counsel for
AEC Regulatory Staff

Enclosure
As stated

cc w/encl, hand-delivered:
Samuel W. Jensch, Esq.
Dr. John C. Geyer
Mr. R. B. Briggs
Leonard M. Trosten, Esq.
J. Bruce MacDonald, Esq.
Paul S. Shemin, Esq.
Angus Macbeth, Esq.

cc w/encl:
Algie A. Wells, Esq.
Mr. Stanley T. Robinson, Jr.
Mr. Daniel R. Muller

16

OFFICE ▶	OGC	OGC	OGC			
SURNAME ▶	MKarman/rmh	JBKnotts	TFEngelhardt			
DATE ▶	5-12-71	5-12-71	5-12-71			

In the Matter of Consolidated Edison Company of New York, Inc.
(Indian Point Nuclear Generating Unit No. 2)
Docket No. 50-247

AEC Regulatory Staff
Responses to Second Round Questions
of the
Citizen's Committee for the Protection of the Environment

A. Responses to Set I (3/9/71)

B. Comments on Set H responses by applicant

C. Responses to Additional Set I (3/15/71)

17

A.

Responses to Set I (3/9/71)

Answer to Question 1, Set I (3/9/71)

1. There is a fundamental difference between the Loss of Fluid Test (LOFT) experimental program and the methods now being used for analysis of Loss-of-Coolant Accidents (LOCAs) for large water-cooled Pressurized Water Reactors (PWRs). The LOFT tests will provide experimental results of overall core and ECCS performance for the specific tests being performed, whereas the methods now being used to predict the consequences of a LOCA are largely analytical supplemented by some experimental work on individual parameters or features. The LOFT experimental program is designed to complement the analytical program in that it will provide a means of verifying results of the analytical efforts. Hence, it is not meaningful to discuss differences in the ability to determine the result of a LOCA from the LOFT program and from the methods now being used for analysis. Rather, both programs are directed toward achieving an understanding of the course and consequences of a LOCA. Both programs are needed from the standpoint that in analytical work experimental verification is often desirable and similarly, to achieve a full understanding of experimental work, analytical vertification is necessary. Taken together, experimental and analytical programs are designed to provide a comprehensive spectrum of information pertinent to a LOCA.

Experimental work on individual parameters or features was mentioned above.

One method to determine the course and consequences of a nuclear power plant LOCA is to use analytical models and techniques developed from theoretical considerations coupled with experimental results obtained from tests designed to investigate a particular feature or phenomenon associated with the accident sequence. Such experimental investigations are commonly referred to as "separate effects tests." As much flexibility as practicable is incorporated into a particular separate effects test program in order to obtain a broad range of information relative to the test objectives. Key parameters are usually varied over a range that extends well beyond that expected to actually occur in the accident in order to examine limiting conditions and to assess safety margins.

Analytical correlations that conservatively predict the phenomena of interest are developed from the experimental data obtained from the separate effects tests. These correlations are computerized, and the resulting computer codes are among the tools used to perform the analytical predictions of the overall accident sequence.

No LOCA has occurred. Thus, there is no record of a LOCA in a water cooled nuclear power plant available to permit comparison of an analytically predicted LOCA sequence with that which would actually occur. The LOFT program is designed to afford a basis for such comparison and to provide information in key areas of the LOCA sequence. It is therefore considered a focal point for water reactor safety investigations. Because the LOFT program will be a series of LOCA tests of an actual PWR, the interrelationship between features of the accident that could not otherwise be obtained from separate effects tests will be demonstrated. The specific mission of the LOFT Integral Test Program is to obtain experimental data that can be used to evaluate analytical methods to analyze and predict:

(a) The coupled thermal-hydraulic-structural response of the reactor primary system during LOCA conditions of large PWRs,

(b) The performance of the emergency core cooling systems (ECCS), the low capacity-high pressure (makeup water), the intermediate pressure, gas-driven accumulator ECCS, and the high capacity, low pressure (flooding) ECCS, and

(c) The margins of safety inherent in the performance of these emergency core cooling systems.

The LOFT program is also designed to provide other pertinent information relative to engineered safety systems during LOCA conditions including:

(a) Containment response,

(b) Containment pressure reduction by spray or other cooling action, and

(c) Identification of any unexpected event or threshold that is exhibited in the response of either the LOFT plant or the LOFT engineered safety systems.

A further objective of the LOFT program is to provide experience in the development and application of standards and codes generally applicable to PWRs by their development and use on LOFT.

With reference to the quotations from AEC Authorizing Legislation, Fiscal Year 1971, Part 3, March 11, 1970, cited in the question, the first and third quotations are responded to by the foregoing. The second, fourth and fifth quotations relate to the Full-Length Emergency Cooling Heat Transfer Tests (FLECHT) programs. In these programs, electrically heated assemblies simulating full-size reactor fuel pins are cooled by sprays and flooding; this is an example of a separate effects test. These tests are designed to examine that part of the LOCA accident sequence following primary system blowdown and associated with the core cooling action of ECCS.

In the PWR-FLECHT tests, ECCS flooding rates, initial fuel pin clad temperatures, and other parameters are varied over a wide range to investigate design and off-design conditions. Flow blockage investigations in which the coolant channel area has been significantly reduced have been performed in the PWR-FLECHT tests. The results indicate that even with flow blockage of up to 90 percent of the channel area, the ECCS is effective. The PWR-FLECHT test results indicate that the reactor core is coolable over a wide range of ECCS flow and initial fuel pin surface conditions. These tests have been designed to be representative of the power densities encountered in current nuclear power plants, including that of Indian Point Unit 2.

Answer to Question A, Set I (3/9/71)

2. In the context of the statement referred to, the term "realistically analyzed" refers to events subsequent to an hypothetical core meltdown accident. The assumptions inherent in postulating the occurrence of core meltdown accidents are that a LOCA has taken place and the emergency core cooling systems fail to cool the core sufficiently to prevent substantial melting. The Regulatory Staff position on this is that core meltdown must be shown to be of such low likelihood that specific safeguards to cope with this condition are not required. We would not recommend licensing a plant if we thought melting of substantially all of the core could occur.

Answer to Question 3, Set I (3/9/71)

3. A list of names of AEC Regulatory Staff personnel who participated in the review of the Indian Point Unit No. 2 will be provided separately.

There is no page 7 or 8

Answer to Question 4, Set I (3/9/71)

4. Two copies of the latest Report of the Advisory Task Force on Power Reactor Emergency Cooling are provided herewith.

Answer to Question 5, Set I (3/9/71)

5. The term "as low as practicable", when used in connection with 10 CFR Part 20, applies to total quantities of radioactive materials released. For Indian Point Unit No. 2, releasing gaseous waste materials from the superheater stack rather than the plant vent would not alter the total quantity of released material. Nevertheless, in theory, some slight reduction in offsite concentrations and ultimately doses is possible by releasing materials from the superheater stack; however, in view of the low dose predicted from gaseous effluents we have concluded that the application has already met our requirements in this regard.

Answer to Question 6, Set I (3/9/71)

6. The selective examination of a licensee's test program is based on a policy of obtaining assurance that each licensee is conducting his testing program in accordance with commitments made in his application. The witnessing of selected tests by Compliance, although considered vital, is only part of our program for obtaining this assurance. Our inspection program includes inspection activities in the following test related areas:

1. An evaluation of the competence of the licensee, and his contractors, to properly conduct the prescribed testing program.
2. Direct observation of the preparations made to conduct the testing program, including procedural development, training of personnel, and the extent and adequacy of the licensee's technical and managerial reviews.
3. Review of selected procedures prior to their use. This review is conducted by a Compliance Regional staff, Compliance Headquarters, other divisions within Regulatory (as appropriate), and consultants (as appropriate).
4. Witnessing the performance of selected tests.
5. Review of selected test results, in addition to the review of the results of tests actually witnessed.

The Division of Compliance has inspected the testing programs for all licensed reactors. The experience gained during these inspections has been, and is being, factored into the design of our inspection program. Such factors include recognition of: Deficiencies detected at other reactors; similarities between tests and test procedures conducted at all reactors; and administrative and organizational deficiencies found at other reactors in preparing for and implementing their test programs.

The selection of tests to be witnessed is based on our evaluation of the safety considerations involved and the degree of confidence obtained by Compliance during our overall inspection program on the capabilities and demonstrated performance of each licensee. Our program of selective examination of tests performed by licensees is considered to be adequate if no unusual situations or conditions are detected. Our program is designed to be intensified to the extent necessary if problems develop.

The licensee has no knowledge of the tests which as a matter of policy, or as part of the inspection plan developed for a particular facility, we have decided to witness. Obviously, however, the scheduling of inspection visits necessitate inquiries by our inspectors as to the particular date for tests planned to be conducted within a known time period. Such inquiries are made shortly in advance of the earliest date of the tests. A number of dates for particular tests are requested both for tests which our inspectors do and do not plan to witness.

The method used in deciding what tests (and how many) to witness is based on internal Compliance instructions, which are not disclosed directly or indirectly to licensees, and on the demonstrated performance of the licensee. During active testing periods, Compliance conducts two inspections per month on the average. An inspection ranges from one to several days.

Answer to Question 7, Set I (3/9/71)

7. The applicant has responded to this question by reference to a non-proprietary version of WCAP-7422. The pertinent Idaho Nuclear Corporation reports will be made available separately.

Answer to Question 8, Set I (3/9/71)

8. In its letter of February 26, 1968, on the Report of the Advisory Task Force on Power Reactor Emergency Cooling, certain concerns were expressed by the ACRS in regard to the preservation of the heat transfer area and coolant flow geometry, design for the hydraulic effects of the loss of coolant accident, testing of the core cooling system at higher temperatures and under degenerated conditions such as core distortion, and the development of programs directed toward gaining better understanding of the phenomena and mechanisms important to the course of a large-scale core meltdown. The objectives and status of the LOFT program in the resolution of these concerns are discussed in the responses to Questions I-1 and D-54. Our position relative to core meltdown is stated in the response to Question I-2. In addition, the Committee expressed the belief that "further research is needed to ascertain the modes of fuel rod failure and to determine that failures will not propagate or tend to block coolant flow excessively." AEC funded research connected with these topics is discussed in WASH-1146, a document previously furnished to the parties and to the board (see page 7 of staff responses to Board questions forwarded with our letter of April 15, 1971).

The Committee called attention to "the need for considering deterioration during the life of the reactor and the role that periodic inspection could play in alleviating this potential difficulty." The regulatory staff and applicants now use Section XI of the Boiler and Pressure Code as a basis for development of periodic inspection programs. This document has been developed subsequent to expression of the concern.

The Committee expressed concern with "the possibility of thermal shock effects on the pressure vessel, or other parts of the primary system, as a result of the rapid introduction of emergency cooling water . . ." and endorsed the Task Force recommendation for "improvements in primary system integrity to reduce still further the already low probability of primary system boundary failure." The additional data needed to resolve the uncertainties in the fracture toughness properties of reactor vessel material are expected to be provided by the Heavy Section Steel Technology (HSST) research and development program. Since reactor vessel materials are initially ductile and their fracture toughness properties are not significantly changed upon irradiation during at least the first five years of operation, the potential for reactor pressure vessel failure as a result of cold water injection is considered to be acceptably small during this period. Sufficient data should be available from the HSST program to permit a final judgement within a five-year period on the acceptability of the projected behavior of vessel material throughout its service lifetime. In the event that the results of the HSST program or other research indicate that the potential for growth of defects in radiation embrittled reactor pressure vessel material reduces the available margin of safety against brittle fracture to an unacceptable level, an engineering solution to the problem could be applied, for example, thermal annealing of the pressure vessel material.

The Committee commented that "deliberate allowance should be made for the possibility of aggravated accident conditions introduced by possible design errors, by weaknesses common to redundant components, or by other unexpected conditions, and full attention should be given to the possible advantage of diverse approaches to the design of emergency core cooling systems" in the evaluation of emergency core cooling systems. The approach to accommodating the possibility of design errors has been to provide excess capacity in the various subsystems. Weaknesses common to redundant components (common mode failures) is a topic presently under investigation in a generic basis with all reactor vendors.

The Committee recommended that "a positive approach be adopted toward studying the workability of protective measures to cope with core meltdown" and noted the Task Force proposal for "study of preventive measures to be made effective prior to loss of containment integrity to minimize the ultimate hazard ...." The regulatory staff has recognized the importance of having a basic understanding of the potential core meltdown accident and has supported various efforts to improve this understanding such as the Emergency Core Cooling Advisory Task Force Study, a copy of which was provided in response to question 4 of this series of questions. However, based on the results of these and other efforts, including those of the reactor vendors, it is our conclusion that the best assurance of public safety is to prevent core meltdown from taking place. We thus have continued to emphasize efforts to assure reactor coolant system integrity and emergency core cooling effectiveness.

Comments on Set H responses by applicant

Question I-9 (3/9/71)

Regulatory Staff comments on applicant's responses to Set H Questions.

Q. 1-2. We have reviewed the applicant's responses and have nothing to add.

Q. 3-4. No comment.

Q. 5-17. We have reviewed the applicant's responses and have nothing to add.

Q. 18. Tornado loads were not considered as a design criterion at the time of the Indian Point 2 review for a construction permit. Subsequently during the operating license review we concluded that, in view of the design features of the plant, backfitting to accommodate tornado wind loads would not be warranted.

Q. 19. No comment.

Q. 20. The ASTM-E6 standard defines steel ductility as the ability of the material to deform plastically (i.e., beyond the yield point) before fracturing.

The applicant's Containment Design Report, Section 3.4.4 indicates that although the liner may yield locally in tension it never yields in compression. Therefore, "ductile behavior under compressive stress" is not a concern at this facility.

Q. 21. We agree that operation of a load following control system would have no effect on the safety features of the plant.

As indicated in the response to FSAR Question 7.3, isolation amplifiers are used in instances where protection system signals are required for other than protective functions. Although this does not represent complete electrical separation, the conclusion that there would be no effect on the safety features resulting from a load following control system malfunction is unchanged.

Q. 22. We have concluded that the design of the flame recombiners is suitable for the Indian Point 2 containment and acceptable as an additional safety feature to increase the level of protection against the consequences of the loss of coolant accident. This conclusion is based on

(a) the absence of an identified mechanism (i) that would produce the conditions specified in the statement of the question (H-22) and (ii) that is not specifically provided for by other safety features in the Indian Point 2 plant,

(b) the design of the recombiners to accommodate the conservative predictions of the course of the loss of coolant accident, and

(c) the addition within two years of the controlled purge system with filters which serves as a backup for the redundant flame recombiners.

Q. 23. We agree with the applicant's statement that the heat of recombination from the recombiner operation is insignificant in terms of the energy discharged during the double-ended rupture of either the cold leg or hot leg. The operation of both recombiners would result in a heat load less than five percent of the decay heat of the core.

Q. 24-25. We have reviewed the applicant's response and have nothing to add.

Q. 26. The staff is aware that development work is proceeding on a catalytic recombiner system. We have not as yet had a catalytic recombiner system presented to us for review and incorporation into a containment. The staff therefore cannot meaningfully comment on the uncertainties associated with a catalytic system.

Q. 27. We have not made a comparable calculation.

Q. 28. We have reviewed the applicants response and have nothing to add.

Q. 29. At the time of the construction permit review for Indian Point Unit 2 no consideration was given to the potential for the crash of an airplane into the containment building.

However, the question of commercial and general aviation overflights and the proximity of airports were a matter of concern in subsequent reviews. In the course of these reviews we have concluded that the hazard from aircraft overflights do not warrant special measures when the facilities are not in the immediate vicinity of airports since statistics available on civilian and general aviation crashes indicated a very low probability of striking any given point near air corridors. We have concluded, however, that the area immediately around airports has a significantly higher crash probability and have under development explicit criteria concerning the design and location of nuclear power plants in relation to nearby airports. Specific consideration has been given to the Indian Point site in this connection and it has been concluded that the distances to major airports from Indian Point are such that the crash probability is not significantly greater than the very low "background" value.

In addition we have recently been concerned with the potential hazard to nuclear power facilities from low level military training flights. On the basis of our review of information on these flights, we conclude that there is no significant hazard to the Indian Point site from such flights.

Q. 30. We have reviewed the plant security measures during the course of the operating license review and found them acceptable. We agree with the applicant's response that sabotage of the conventional parts of the plant would not cause any radiation hazard to the public. Sabotage of the nuclear portions of the plant by individuals not intimately familiar with the design and operation of the reactor is protected against by the redundant nature of the protection systems.

Q. 31. The reactor containment provides the primary protection against external missiles. Explosive charges were not specifically considered by the Staff in its review. Applicants are not required to provide for design features or other measures for the specific purpose of protection against attacks and destructive acts directed against the facility by an enemy of the United States, whether a foreign government or other person (10 CFR 50.13).

Q. 32-33. We have reviewed the applicant's responses and have nothing to add.

Q. 34. The AEC requirements for reporting releases of radioactivity are met by Sections 3.9.A.3 and 6.6 of the Technical Specifications.

Q. 35-37. We have reviewed the applicant's responses and have nothing to add.

Q. 38. Total failure of the ECCS as postulated in Part (a) has not been evaluated as noted in our response to Question 2 in A above. We have nothing to add to the applicant's response to Part (b).

Q. 39. The staff has evaluated the charcoal adsorbers in terms of a minimum 10% removal effectiveness for organic iodides per pass over the entire period of operation. This analysis is based on the most adverse results reported in ORNL-TM-2728, where very low efficiencies were reported for charcoal beds after flooding. The staff does not expect flooding to occur over an extended period, but the possibility of such occurrence cannot be placed at zero.

Q. 40. The safety systems discussed in the applicant's response were evaluated in the course of the operating license review and found to be acceptable.

Q. 41. We have not made a comparable evaluation and have no basis for comment. The protection system for Indian Point 2 was accepted on the basis of evaluation against the single failure criterion.

Q. 42. Extensive tests on the iodine removal capability of containment spray systems have been completed at Oak Ridge National Laboratory and at Battelle Northwest Laboratory. The scaling factors were of the order of 1:5000 and 1:100 of full size, respectively. All results indicated rapid iodine removal and could be correlated with theoretical calculations. Therefore, the current staff model for the evaluation of spray effectiveness, which incorporates a factor of conservatism of greater than three in the iodine removal constant, tends to considerably underpredict the expected minimum performance of sprays. The staff is convinced that the effectiveness tests have given adequate assurance of the performance characteristics of spray systems and have defined the degree of conservatism of the current staff model.

Iodine removal by sprays and by plateout are both realistically taken as time-dependent mechanisms, and are therefore interrelated. The considerably slower rate of iodine removal by sprays predicted by the staff model would therefore permit greater plateout than the much more rapid spray removal rate chosen by the applicant. A comparison of a realistic performance model, based on simultaneous plateout and spray removal, with the model based on the TID-14844 plateout assumptions and the same spray removal performance showed that the latter model (as presently applied by the staff) yields the more conservative results.

Q. 43. We have reviewed the applicant's response and have nothing to add.

Q. 44. We have nothing to add at this time. See our letter to the Board dated April 29, 1971.

Q. 45. As indicated in the applicant's response, analyses have been performed assuming the design leak rate for the duration of the accident. It is these analyses upon which the Staff evaluation was made and the design accepted. No credit was given for operation of the isolation valve seal water system and the penetration pressurization system in the Staff's evaluation of the consequences of the accident.

Q. 46. We have reviewed the applicant's response and have nothing to add.

Q. 47. In calculating the effects of phenomena other than fission product release from the core, such as hydrogen generation in the containment, the Staff uses the most conservative model characteristic of each phenomenon, which may or may not be consistent with the assumption stated in TID-14844.

We find the applicant's approach in this instance to be acceptable.

Q. 48. No comment.

Q. 49. No comment.

Q. 50-54. We have reviewed the applicant's responses and have nothing to add.

Q. 55. The staff has stated its position in the response to Question A-44.

Q. 56.(d) In the context of any scientific or engineering endeavor absolute 100% certainty cannot be assured. There is always some finite, albeit small, likelihood of the unlikely happening. With reference to ECCS function, we are continuing to evaluate the status of present knowledge related to ECCS effectiveness so that we can be assured that the likelihood of failure is sufficiently small as to be negligible.

Q. 57. No comment.

C.

Responses to Additional Set I (3/15/71)

Answer to first Additional Question, Set I (3/15/71)

1. A copy of the U. S. AEC Press Release H-262 (December 3, 1965) with the attached letter from W. D. Manly is provided herewith.

Answer to second Additional Question, Set I (3/15/71)

2. The analysis requested has been sent to the AEC under cover of a letter dated March 5, 1971 from Westinghouse in the form of a proprietary report entitled "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors." A request for such a report should be made in accordance with 10 CFR 2.744.

Answer to third Additional Question, Set I (3/15/71)

3. The applicant has responded to this question by reference to WCAP-7561 in connection with the response to Question H-11(b). This report identifies the pertinent data in a convenient form.

ET NUMBER
PROD. & UTIL. FAC. 50-247

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. H-262
Tel. 973-3335 or
973-3446

FOR IMMEDIATE RELEASE
(Friday, December 3, 1965)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MAKES RECOMMENDATIONS ON NUCLEAR POWER PLANT DESIGN

The Atomic Energy Commission has received a report from its Advisory Committee on Reactor Safeguards outlining the Committee's recommendations concerning pressure vessels and engineered safeguards for pressurized water and boiling water nuclear power plants.

A copy of the Committee's letter together with a statement by the Commission on the safety research program is attached.

#

12/3/65

STATEMENT BY AEC ON
SAFETY RESEARCH PROGRAM

The Atomic Energy Commission has received a report from its Advisory Committee on Reactor Safeguards which comments on many improvements made in pressurized and boiling water reactors with consequent marked reduction in the risk of significant radiation exposure to the public. The report then outlines certain technical areas relating to design, construction and operating surveillance of pressure vessels for water reactors on which the committee recommends that additional work be done.

The ACRS reconfirms its belief that no undue hazard to the health and safety of the public exists in water-type reactors. However, the committee notes that "orderly growth of the industry" is occurring, with increase in the number, size, power level and proximity of reactors to large population centers, which will in the future make it desirable and prudent to incorporate in many reactors the design approaches which it recommends be developed. Inspection techniques, stress analyses, flaw propagation during vessel use, protection of the containment against missiles (equipment parts or fragments) and pressure in the unlikely case of vessel failure, and other related matters are recommended by the ACRS for increased attention.

The Commission welcomes these specific recommendations of the ACRS as being in keeping with and supplementary to statements made by the Commission to the Congressional Joint Committee on Atomic Energy in June of this year. At that time the AEC noted that the rapid expansion and development of the nuclear power industry, including increased size of proposed plants and the incentives to locate in closer proximity to metropolitan load centers, have re-emphasized the continuing need for careful attention to all matters which potentially could affect the health and safety of the public.

The Commission also stated in its testimony that further important advances in reactor plant design, in the capability of safety systems and engineered safeguards, in adapting critical components and systems to accommodate their inspection and testability, and in practical demonstration of dependability of performance of such critical systems, must evolve to keep pace with the development of the nuclear

(more)

power industry. It also was stated that an augmented and reoriented safety research and development program would be undertaken by the Commission, in collaboration with industry, to accomplish the improvements which would be required to keep pace with the developments in reactor technology.

The recommendations of the ACRS on pressure vessels will be given prompt attention by the Safety Research Steering Committee in its review and development of the augmented safety research and development programs. The Commission envisages that steady improvements in safety technology will evolve from these augmented and accelerated programs and that the safety of reactors will continue to advance. In the meantime, as both the AEC and the ACRS have stated, the adequacy of safety provisions in each reactor will continue to be established by thorough and detailed analysis and evaluation on a case-by-case basis.