UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

3-10-72

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

RESPONSE OF THE AEC REGULATORY STAFF TO PROPOSED FINDINGS OF FACT OF CITIZENS COMMITTEE FOR THE PROTECTION OF THE ENVIRONMENT

I. Introduction

On February 8, 1972, the Citizens Committee for the Protection of the Environment (Citizens Committee) filed its Proposed Findings of Fact and Memorandum of Law relative to the motion of the applicant for authorization to test at up to 50% power for the Indian Point Nuclear Generating Unit No. 2. The regulatory staff on the same date filed its comments on the applicant's proposed findings. The staff will respond to those portions of the Citizens Committee's findings which we feel are erroneous or need further clarification. It is not feasible to respond to all the comments and non-evidentiary matters included within the findings. We will respond to the Citizens Committee Memorandum of Law as a separate item in this response.

II. Responses to Proposed Findings of Citizens Committee

1.(b. & c.) These proposed findings make reference to TID-14844 assumptions and WASH 740 analysis. The staff comments relative to these reports were made in responses to Board questions at the hearing session of January 19, 1971. These responses were submitted on April 15, 1971, and were subsequently accepted by the Board as evidence in this proceeding. The staff conclusions were that WASH 740 was not a relevant document, and the assumptions in TID-14844 applied to release into the containment (rather than directly into the atmosphere as implied in proposed finding 1(b)) and are conservative assumptions for site evaluation purposes. Our position remains the same.

1.(e.) The consequences of a major meltdown have not been evaluated for Indian Point Unit 2 since a major meltdown is not considered to be a credible event. No analysis of a major meltdown for a contained reactor which would describe the likely consequences has been made (Tr. 3978, 3979, 3983).

2. The staff position is that if the emergency core cooling system functions as designed, the release of fission products from the core to the containment building would be minimal, since very few pins would sustain clad damage, and release from these is limited to gap and plenum inventory. (FSAR 14.3.5 - Question 14.1)

3. The staff has not only estimated maximum probable derivations for all parameters entering into the calculation of the spray system iodine removal effectiveness, but has added all factors of uncertainty in such a way that all combine to give the most conservative spray removal value.

Factors such as testing of nozzles singly or at heights different from those in the containment are immaterial in establishing the drop size spectrum and trajectory. The results can be extrapolated analytically to the realistic situation. The physical characteristics of the proposed spray solution are virtually identical to pure water, so that the results are identical. Nozzles were tested for the minimum pressure drop expected as well as for higher pressure drops.

The staff did acknowledge uncertainty in predicting the exact drop size spectrum, but the value of 2,000 microns chosen for the staff model is conservative on the basis of all published data for the nozzles used (Tr. 1494). This is substantiated by iodine removal test data in the ORNL and BNWL facilities with these nozzles.

Turbulent mixing would produce greater uniformity and more rapid averaged iodine reduction. The general variations of iodine concentration of up to 20% in different volume elements would be diminished by rapid mixing.

The elemental iodine removal effectiveness of the sprays is evaluated for the highest predicted containment temperature of 280°F, which yields the most conservative result (Tr. 1528-1534).

Calculations of drop coalescense are based on a model which takes into account drop trajectories. A hypothetical case of segregation into streams of particles of discrete size crossing at right angles has no applicability to the actual situation (Tr. 1507).

Calculations of drop residence time are based on peak predicted containment temperature and pressure, and a steam-air atmosphere. Turbulence would, in general, result in longer suspension times and more favorable iodine removal conditions (Tr. 1507).

The effect of steam condensation adds only a very thin surface layer to the drop, and mixing and diffusion would quickly equalize the distribution of the reactive additive. The positive net effect of iodine transport to the drop surface by the condensing steam was not included in the staff analysis.

Both sodium hydroxide and alkaline thiosulfate would give iodine removal factors sufficiently large to satisfy the requirement of Part 100. The choice between reagents was made by the applicant (Tr. 1625-31, 1634-35).

The dose calculations for Indian Point 3 were performed on the basis of TID-14844 guidelines. Plateout is assumed to reduce the iodine reaching the containment by a factor of two. Comparison with a proposed realistic model showed this assumption to be conservative.

3.(b.) The proposed finding implies that the staff assumption for charcoal filter efficiency was not conservative. As indicated in the testimony, the use of a filter efficiency of 10% was characterized as "superconservative" (Tr. 1305, line 1) and "realistically conservative" (Tr. 1308, line 18). This conservatism is in fact admitted by the Citizens Committee in proposed finding 11(g)(2). The filters are operative during the first two hours but contribute relatively little to iodine removal effectiveness in comparison to the much more rapid action of the sprays. Their contribution is not required to an extent greater than the assumed minimum value.

The staff analysis for the Indian Point Unit 3 plant did not assign a specific filter efficiency, but only stated that it was anticipated that a minimum efficiency of 5% per pass was attainable (Tr. 1300, lines 6-10).

3.(c.)(1) & (2) Proposed finding 3(c) implies that the recombiners could not be operated in the time period required to keep the hydrogen concentration below the flammable limit. Applicant's calculations show that the containment pressure is reduced to 5 psig at about 3,000 seconds or less than one hour (FFDSAR Fig. 14.3.4-2). Applicant's calculations show the time period required to reach a flammable hydrogen concentration is about 80 days (Fig. 14.8-1 in answer to staff question 14.8, Volume 6 FFDSAR).

3.(c.)(3) The record indicates only a possible minimal effect (Tr. 2279).

3.(d.) The staff has not given any credit for this system or the penetration pressurization system in the evaluation of the doses associated with the loss-of-coolant accident (Staff Safety Evaluation p. 62).

The allowable test leakage rate as provided for the containment integrated leakage rate test is only 0.075% per day if the test is run at accident pressure. It is even less than this if the test pressure is below accident pressure as provided for in the Technical Specifications (see specification 4.4 II.A.5. on page 4.4-3).

4. The staff has explained its position on the percentage of methyl iodide which should be assumed as a component of the containment atmosphere following a LOCA, and which are given in Safety Guide No. 3. The fractional

during the blowdown and reflood portions of the accident. For example, the Westinghouse Evaluation Model considers 80% of the predicted blowdown flow rate through the core for calculating fuel rod heat up in the hot spot of the core. While the staff considers this assumption to be conservative with regard to potential flow maldistribution through the core for the entire period of the blowdown phase of the accident, it provides additional margin for potential flow redistribution which may occur as a consequence of fuel rod swelling during a portion of the blowdown phase. Additionally, the Westinghouse Evaluation Model does consider changes in the heat transfer coefficient in the gap separating the fuel pellet from the cladding. During the reflood portion of the accident, potential changes in core geometry which include local as well as core average reductions in coolant channel flow area have been considered by the staff in assessing heat transfer characteristics of distorted geometries. Distorted flow configurations have been tested in the FLECHT program, WCAP-7665 as well as by the Aerojet Nuclear Corporation, (Tr. 3514).

5.(c.)(1) The response to 5.(b.) is applicable to the proposed finding.

5.(c.)(2) The experiments performed by ORNL and most recently reported in ORNL-4752 have shown that flow area reductions have occurred at least over a 9-inch length of core height (Table 8 of ORNL contains a list of blockage measurement for this range of rod length for three PWR multi rod burst tests). Since the length of swelling along the length of rod is usually limited to 1 to 1-1/2 inches, randomness of swelling has been demonstrated by these tests. Westinghouse Experiments performed by Westinghouse have shown similar results. (Tr 2510 through 2518)

5.(d.) The documents cited are not in evidence in this proceeding.

5.(e.)(1)(2)(3) The documents cited are not in evidence in this proceeding.

5.(b.)(1)&(2) The documents cited are not in evidence in this proceeding.

5.(g.) The documents cited are not in evidence in this proceeding.

5.(h.) The document cited is not in evidence.

5.(i.)(1)(2) The documents cited are not in evidence.

5.(j.)(1), (2) & (3) Quenching can cause fuel rod failure. The notion that temperature rise rate is an important parameter in studying brittle failure is not generally accepted. The integrated effect of time at temperature in determining the extent of α + oxide penetration or the amount of zirc-water reaction is the more meaningful determinant. Only insofar as a temperature rise implies a time at various temperatures is it a factor to be considered (Tr 2187-90).

5.(k.)(1) The single rod fuel burst tests have application with regard to showing sensitivity of the various parameters with regard to the degree of swelling and burst characteristics. Since single rod burst tests have shown that interaction between neighboring rods will occur, they cannot by themselves be used to infer degrees of coolant channel flow area reduction in PWR multi-rod geometries. As stated in the response to CCPE Proposed Finding 5.d., multi-rod burst experiments are the only quantitative measure for determining coolant channel flow area configuration in an open lattice PWR geometry.

5.(k.)(2), 5.(k.)(2)(a), 5.(k.)(2)(b), 5.(k.)(2)(c), 5.(k.)(2)(d), 5.(k.)(2)(e) The documents listed are not in evidence and findings are not supported by evidence.

5.(k.)(3) The tests were performed to compare swelling and burst characteristics between irradiated and unirradiated test specimens. Dimensional differences between these rods and those installed in Westinghouse PWR's have no bearing on the phenomena observed.

5.(k.)(3)(b) Dimensional differences in rods will effect burst characteristics and can be related in a quantitative manner from hoop stress considerations. The results of these experiments are not relevant to coolant channel flow area reductions resulting from swelling for a Westinghouse designed PWR.

5.(k.)(3)(c) The experiments performed by ORNL and others have shown that irradiated rods swell less than unirradiated rods for similar test conditions. In ORNL-TM-3636, page 1 of the Abstract, it states, "However, irradiation effects in the Zircaloy cladding appear to reduce the extent of expansion."

5.(k.)(4)(a)A., 5.(k.)(4)(a)B. The Staff response to CCPE Proposed Finding 5.(k.)(2) applies.

5.(k.)(4)(b)A. & 5.(k.)(4)(b)B. Document listed is not in evidence.

5.(k.)(4)(d) The Staff response to CCPE Proposed Finding 5.(c.)(2) applies.

5.(1.)(1) The document listed is not in evidence.

5.(1.)(2) The Staff response to CCPE Proposed Finding 5.(d.) applies.

5.(1.)(3) The document listed is not in evidence.

5.(m.)(1), 5.(m.)(2), 5.(m.)(3) The significance of the PWR flow blockage tests must be placed in context for proper understanding. The performance of a fractional-blockage geometry, e.g., plate, sleeve, or "realistic," becomes important only when the results of a totally-blocked geometry becomes unacceptable. For example, consider FLECHT test numbered; 6948, 7946, 8162, 8366, 8668 described in WCAP-7665. All five of these tests were conducted at a nominal initial peak clad temperature of 1600°F, a nominal flooding rate of 1"/sec, nominal inlet coolant temperature of 150°F, peak power of 1.24 kw/ft, and pressure of 58 psia. For test 8668, the highest clad temperature measured along the length of any rod was 2,052°F (at the 8-foot elevation). For this run, 16 of the interior flow channels were completely blocked. For similar conditions with no blockage, test number 6948, the peak clad temperature was 2,067°F, at the 6-foot elevation (bundle midplane). The temperature data for the three tests with blockage between 75% and 100% (7946, 8162, and 8366) were bounded by the two tests cited. Thus it appears that consideration of the results of 100% channel blockage overshadow the lesser effects of lesser blockage. The Regulatory Staff has testified that blockage reduces clad temperature

(TR 3467) and that real blockage gives increased heat transfer (TR 3513).

5.(n.) The document listed is not in evidence.

5.(o.) The Water Reactor Safety Research Program (WASH-1146, not in evidence) does not include the Westinghouse fuel rod failure tests. Most of the ORNL rod failure work post-dates the other 1970 document referenced by CCPE (IN-1382, not in evidence); the latter alleges inadequate understanding of fuel rod failure.

(5.(p.) The net result of the PWR-FLECHT tests with blockage was a demonstrated decrease in peak clad temperature with respect to identical, but unblocked tests. Therefore, the ECCS Interim Criteria evaluation models require no additional conservatism to account for flow blockage of the magnitude predicted for this plant. The 70-100^{o}F temperature rise referenced at TR 2734 is a calculated temperature rise for the moist geometry observed in the Westinghouse multi-rod burst test. Such a temperature rise was not confirmed by the FLECHT tests.

6.(a.)(b.)(c.)(d.)(e.)(f.)(g.)(h.) Document listed is not in evidence.

6.(i.) Calculations indicate that the clad temperature at the end of blowdown is relatively insensitive to the flow maldistribution factor (Tr 3652). If the amount of maldistribution were doubled, i.e. if the factor was lowered from 0.8 to 0.6, the peak clad temperature would increase by only about 100^{o}F.

7. The Westinghouse blowdown code BLODWN has been verified by many experiments. These include several years of LOFT support experiments and blowdown tests performed at Battelle Northwest Laboratories in the Containment Systems Experiment (CSE).

7.(a.)(1) Experimental verification of a blowdown code means assessing its ability to predict key variables in blowdown tests. These tests are conducted with scale model reactor systems which simulate the geometrical complexity of typical reactor internals (Tr 2769-2771). Two such key variables are core axial pressure drop and core flow direction. The CSE and LOFT semi-scale tests have verified that BLODWN can be used to predict these variables. Thus, the pressure drop and flow direction calculated by BLODWN for Indian Point 2 can be and was used to determine the forces on fuel rods (Tr 2752-2753). The springs which hold the Indian Point 2 fuel rods in place were then determined to be of sufficient strength to withstand the predicted blowdown loads (Tr 2753).

7.(a.)(2) The 1/7 scale tests referred to at Tr. 2801-2802 are the same tests referred to at Tr. 2778. These were steady-state flow redistribution tests used for verification of the THINC computer program and therefore have no relation to blowdown load calculations with the BLODWN computer program.

7.(a.)(3) In context, the applicant stated at Tr. 2775 that the Idaho semi-scale tests were not, by themselves, adequate to demonstrate the validity of the BLODWN computer program. The Idaho semi-scale tests taken with the CSE experiments and the pipe blowdown experiments at Illinois Institute of Technology (Tr. 2779) were judged by the applicant to satisfactorily demonstrate the reliability of BLODWN (Tr 2775 lines 11-16).

7.(b.) The 2230 pounds of force referred to at Tr. 2767 is apparently a transcript recording error and should read "20 to 30 pounds"(Tr. 2757 line 17).

7.(b.)(1) Ricochet forces defined by CCPE at Tr. 2757 are in reality form drag forces on rod springs or spacers. These forces were considered in the blowdown force analysis (Tr. 2757, lines 10-14).

7.(b.)(2) The time period when blowdown loads are sufficient to provide a possible mechanism for dislodging fuel elements is the first 50 milliseconds following initiation of a LOCA. In this very short time there would be no temperature change in the rods or springs relaive to their steady-state operating temperatures. Therefore, the applicant has properly ignored differential expansion in assessing the holding capability of the springs under blowdown loads (Tr. 2761, lines 19-24).

7.(c.) The CCPE postulated (Tr. 2875) a non-mechanistic lengthening of the blowdown time by suggesting a LOCA with a variable break area which started large and grew small during blowdown. The resulting increase or decrease in peak clad temperature would depend upon the area vs. time relationship chosen for such an accident. The applicant considered a spectrum of constant area breaks in the FSAR. Variable area breaks are not postulated by the AEC for accident analyses per the ECCS Interim Criteria, and the applicant has not analyzed such an accident (Tr. 2875, lines 15-16).

8.(a) The applicant has demonstrated that the ECCS for Indian Point 2 meets the AEC's ECCS Interim Criteria. Therefore, the core is protected from a major melt-down. The documents cited by CCPE in alleging that ECCS designs are still in the experimental proof stage were dated 1967 and 1970. More recent research and development have added significantly to the proof of the conservatism in the ECCS Interim Criteria.

8.(b.) Core disassembly will not be caused by the amount of clad oxidation (shattering) or bursting predicted by the applicant using the LOCA model specified in the ECCS Interim Criteria.

8.(c.)-8.(d.) The Applicant has provided adequate assurance that rupture of the reactor vessel will not occur. The plant systems are not designed to control the consequences of a core meltdown resulting from such an accident.

9.(a.)(b.)(c.)(d.) The definition of loss-of coolant accident as given in 10 CFR Part 50 limits the size of the pressure vessel rupture to the double-ended break of the largest pipe of the reactor coolant system. This definition was applied by the staff in its evaluation of the proposed ECCS against General Design Criterion 35 and also against the acceptance criteria described in the Commission's Interim Policy Statement. Our conclusion, as stated i- Supplement No. 3 of the staff safety evaluation was that the acceptance criteria could be met.

The rupture of the reactor vessel is not considered a likely event and therefore excluded from the category of accidents known as loss of coolant accidents for the following reasons:

> The probability of failure of a reactor vessel built in accordance with the rules of construction code (ASME Section III - Nuclear Vessels) specifically formulated to provide increased reliability and safety over vessels built to non-nuclear vessel codes (ASME Section I - Power Boilers) is considered negligible. The basis for this conclusion is supported by the statements contained in "Report

by AEC Regulatory Staff in Response to ASLB Questions Concerning Reactor Vessel Integrity" (dated October 26, 1971). The Safety of fossil-fueled power boilers in the U. S. built to ASME Section I code rules which are substantially less stringent in the design and construction requirements than those of ASME Section III Nuclear Vessel Code has been demonstrated by successful and reliable operation in many power plants since the development of the first ASME Boiler and Pressure Vessel Code. Nuclear reactor vessels can be expected to exceed the service reliability of power boilers by virtue of the much more demanding requirements imposed by the nuclear vessel code and the unparalleled inservice inspections which reactor vessels will receive during their service lifetime. (Staff Response to Board Questions 10/26/71)

9.(e.)&(f.) The design and construction requirements of the ASME Section III Code, 1965 edition with 1965 Summer Addenda and Code Cases, contain all of the principal rules which appear in later editions of the code. Later editions of the code expand primarily in the areas of quality assurance provisions. Augmentation of the Code, in the case of the Indian Point 2 reactor vessel, by Westinghouse equipment specification, is interpreted as added quality assurance measures responsive to the ACRS recommendations "to give further attention to the methods of analysis, and to the development and implementation of improved methods of inspection." As stated in the above-mentioned "Report by AEC Regulatory Staff," the stress analysis of the reactor vessel received an additional review to verify the adequacy of the methods of analysis employed by the vessel manufacturer and the extent of nondestructive examination applied to the vessel materials

100% ultrasonic inspection) represented an improvement over the lesser requirements contained in the 1965 edition of ASME Code. The inservice inspection program as identified in the Technical Specification for the Indian Point 2 reactor vessel, is subject to review by the AEC upon completion of the first inspection. This procedure which is outlined in the Commission document entitled "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection" (January 31, 1969) is intended to re-examine the feasibility of using newly developed examination equipment which the industry is making available to nuclear power plants for the specific purpose of augmenting the inspections of areas with limited access.

The staff answers to Board questions (Tr. 758-759) received in evidence on July 13, 1971 reflected the fact that the authors of Section XI of the Boiler and Pressure Vessel Code recognize the possibility of continuing advances in inservice inspection techniques. It is not the intent of the Code to limit the period of issuance of a nuclear reactor license on the basis of requirements that may not be capable of being met. As stated on page 31 of the additional testimony of the regulatory staff concerning reactor vessel integrity dated October 26, 1971,

> "The regulatory staff has received assurance from industry that the examination equipment for remote inspections can be made available on a timely basis and applied to satisfy the examination requirements of the ASME Section XI Code, within this five-year period."

9.(g.) Although the fracture toughness properties of the Indian Point 2 reactor vessel materials are not completely available, the AEC regulatory staff has not relied upon this limited data to establish safe reactor vessel operating limits. The approach taken, in this case, was to assume very conservative values of fracture toughness properties as established from the review of many other applications where adequate data were submitted, and to apply the latest AEC fracture toughness criteria (Proposed Amendment to 10 CFR Part 50, dated April 6, 1971), as amended to reflect the recently revised ASME Code Section III fracture toughness rules. This procedure will assure that an adequate margin is available during the startup and shutdown of the reactor vessel to assure operation within pressure-temperature limits where the materials exhibit acceptable fracture toughness properties. A more conservative temperature limit of pressurization for the Indian Point 2 reactor vessel has been established which is significantly higher than for cases where more complete fracture toughness properties are made available.

9.(i.) The H. B. Robinson incident is related to a piping component failure. Similarly, the through-wall piping cracks referenced in (i.)(1) were not cracks in the reactor vessel, but occurred in piping components beyond the reactor vessel pressure-retaining membrane boundary.

It is precisely demonstrated from such reported experiences, of failures in piping components, that the higher likelihood of ruptures can be expected to occur in piping components rather than in the vessel proper. Piping components, by virtue of their geometry are subjected to combinations of loads not experienced by vessels, and as such, are considered as the more likely loss-of-coolant design basis breaks.

The AEC Regulatory staff's evaluation of the H. B. Robinson incident does not conclude that the failure resulted despite compliance with the code requirements, as inferred by paragraph (i.). The design, in the opinion of the regulatory staff, did not comply fully with the code requirements.

10.(a.) The regulatory staff has reviewed changes favorably in a letter to the applicant dated February 25, 1972. This letter was forwarded to the Board and parties on February 28, 1972.

10.(b.) The Compliance Division of the regulatory staff is following the detailed restoration work and will provide findings on the adequacy of such restoration in their determination that the plant is constructed in accordance with the application.

11.(d.) With respect to this item, the staff response to the Board's question (Tr. 500) dated January 19, 1971 stated that "...the purging system is considered to be a backup to the redundant flame recombiners." The decision to permit a two-year delay was based on the staff's best judgment of the situation.

11.(f.)(1) There is nothing in the record or elsewhere which states that a core meltdown was considered credible in 1965. The PSAR requirement for the crucible was based on a possibility of partial fuel melting for a LOCA (Tr. P-1148-1149). No performance tests under LOCA conditions have been run for Indian Point Unit 2 accumulators. Testimony references are to semiscale tests.

11.(f.)(2) Staff testimony presented on pp. 1148 and 1149 of the transcript state that a crucible was required because analysis indicated the possibility of partial fuel melting (not core meltdown). The crucible removal was based on the revised emergency core cooling system, which prevents fuel melting.

12.(a.)(1)(a) This question responded to in Section 14 3.3 of FFDSAR.

12.(a.)(1)(b) Evaluation models approved by the Commission in Appendix A of the Interim Policy Statement do accommodate certain possible changes in core geometry. The staff response to CCPE Proposed Find 5.(b.) applies.

12.(a.)(1)(c) In a reactor system having multiple, similar loops, it is possible to simulate the unbroken loops as one loop containing the total mass and providing the total flow and heat transfer. This is analogous to representing a parallel electrical network by its equivalent single branch. The broken loop is modeled as a separate loop. Thus, a two-loop code would be satisfactory. (See applicant's additional testimony, July 13, 1971.)

12.(a.)(2) As part of a special sensitivity study the effects of a detailed pressure distribution on the natural circulation flows within the core were performed (see Appendix 14.B. of Indian Point 2 FFDSAR).

12.(a.)(3) Intervenor has incorrectly interpreted material presented in the cited reference which is not in evidence.

12.(a.)(4) Exhibit M-136 refers to assumptions and simplifications made regarding blowdown and heatup codes. In reviewing proposed evaluation models (which considered both blowdown and heatup codes) the staff required sensitivity studies to be performed to insure that the level of detail in representing the primary system as well as the hot rod was sufficient and any finer representation of the system produced insignificant differences in predicted blowdown behavior. (See Interim Policy Statement, Appendix A, Part 3.)

12.(a.)(5) The SATAN-V and LOCTA R-2 codes are the two computer programs used for the analysis of LOCA. The SATAN-V code has been checked against the Loft semi-scale and the CSE tests for a number of years. The analytical predictions of the SATAN-V code have been verified for most of the blowdown transient. Where the code did not satisfactorily predict the experimental result, e.g., during accumulator injection, conservative assumptions are applied as required by the Interim Policy Statement. (Interim Policy Statement Appendix A, Part 3.)

The LOCTA R-2 code is used to calculate the fuel element temperature transients during the LOCA. This code is a computer formulation of the basic heat transfer equations, such as the Fourier heat conduction equation, that have been verified for many years. The heat transfer correlations used in this code have been derived from experimental data. Where uncertainties exist, such as "time to DNB," conservative assumptions are applied as required by the Interim Policy Statement. (Interim Policy Statement, Appendix A, Part 3.)

12.(a.)(6) As discussed in number 12.(a.)(5) above, a number of analyses have been made to verify the codes with experimental results. In those areas where some uncertainty exists, conservative assumptions are applied as required by the Interim Policy Statement.

12.(a.)(7) The listed document is not in evidence in this proceeding.

12.(b.)(1) While assumption is based upon facts not in evidence in this hearing, it is normal procedure to perform tests over a range of parameters which bracket the anticipated conditions for which the variable is being considered. For example, rod bursts were performed over a range of pressures, heating rate, and material properties (irradiated versus unirradiated) rather than the specific predicted condition which may occur during a LOCA. The purpose of performing these tests over a range of parameters is used to establish the sensitivity of a given parameter to the variable being studied.

12.(b.)(2) The Staff response to CCPE Proposed Finding 12.(b.(1) applies.

13. The staff always follows the guidelines of TID-14844 in its analyses, and has accepted the design of this plant on the basis of these analyses. In certain instances, the applicant performed analyses which were not the same as those of the staff. For example, one of the applicant's analyses of the environmental consequences of a loss-of-coolant accident (FFDSAR 14.3.5) assumes that the isolation valve seal water system and the penetration pressure system operate in such a way that the containment is isolated in one minute. The staff acceptance of the design was based, however, on the analysis assuming that these systems did not function properly. The resultant doses were within the guidelines of 10 CFR 100 in either case.

The staff evaluation and acceptance of the containment spray system is based on a similar comparison of a model based on TID-14844 assumptions and a model used by the applicant (see staff answers to questions H-42, H-45 - Citizens Committee Exhibit I).

14. The staff denies the premise of the Citizens Committee for the reasons set forth below:

14.(a.)(1) The statement apparently refers to advance instructions for the public. Mr. Davies' testimony does provide evidence of notification to the public regarding specific actions to be taken in the event of an accident (Supplementary testimony of Sherwood Davies, pages 5-6, 9-10 following Tr. 1754).

14.(a.)(2)&(3) The Citizens Committee allegations regarding inadequacy of the state's plans for coping with emergency are controverted by testimony of the state's witnesses, when that testimony is taken as a whole. The state testimony clearly establishes that the state has made a conscious decision to design their response specifically to the conditions that might prevail in a post-accident situation. The arguments of Citizens Committee do not support the contention that the state's approach is inadequate.

14.(a.)(4) The emergency plan for the State of Vermont as described in the transcript of another proceeding, is not at issue in this proceeding.

14.(a.)(5) The state authorities made a decision regarding the use of advance publicity that the Citizens Committee does not accept. The decision, however, is justified on the basis of what is generally understood with respect to poor retention by the public of information concerning alert and warning signals for an enemy attack.

14.(a.)(5)(b) Contrary to the implication of the Citizens Committee, evacuation may not be the most desirable protective measure following an accident. The state emphasis has been placed on assessing each accident situation on an ad hoc basis inspection requirement.

14.(b.) The testimony cited by Citizens Committee points up the staff's conclusion that the requirements we have placed on the applicant regarding those aspects of his proposed security plan to be implemented prior to criticality meet the requirements of an adequate security program. Other aspects of the security plan designated for later implementation enhance the plan, but do no more than make an adequate program better.

15. The applicant's testimony, as a whole, in the In Camera hearing, supports a finding of adequacy of the security plan for this facility.

16. There is no testimony or evidence to support this finding.

17.(a.)&(c.) Our previous responses to proposed finding 9, 11(d.) are applicable to the contentions related to safety features and are set forth in the record (Tr. 1879-1882).

17.(a.)(2) & (b.)(2) See staff response above to ECCS matters.

17.(b.)(1) & (c.)(2) We believe this contention refers to the staff's informal review of the state emergency plan, which is dealt with in our response 14.

17.(b.)(2) The staff denies any contention that documents were withheld from the Citizens Committee.

18.(a.)(1) &(b.)(1) The staff denies that its review of the state plans was "cursory." The testimony indicates that the word used by the staff witness was "informal," and the reasons for the informal review to determine the most desirable response. Although one possible acceptable approach to planning for emergencies involved preplanned evacuation of the low population zone, the state's approach is also a valid basis for planning, and does provide for more flexibility in response.

19.(a.)(1)(a), 19.(a.(1)(b), 19.(a.)(1)(c), 19.(a.)(1)(d), 19.(a.)(1)(e) - The Westinghouse evaluation of the consequences of a LOCA occurring while operating at 50% of rated power indicated that peak clad temperature would be less than 1200^{o}F and for the intended duration of the test period, the internal gas pressures would not exceed 100 psi. (Tr. 4033.)

19.(b.)(1) There is no evidence to support the premise that ECCS equipment will fail to perform within their requirements specified by design. On the contrary, tests are periodically performed to insure that all ECCS equipment meet their functional requirements.

19.(b.)(2) See reply to CCPE Proposed Finding 19.(b.)(1).

19.(b.)(3) See reply to CCPE Proposed Finding 19.(b.)(1).

20. The regulatory staff complied with the requirements of Section D.2. of Appendix D to CFR Part 50 in its Discussion and Conclusions, dated December 30, 1971.

With respect to intervenor's Proposed Conclusions of Law, the regulatory staff's position is that, for the subject motion before the Board, namely a request for 50% power testing, the Applicant has proven that the five preprequisities cited by intervenors have been met, and the 50% testing license should be authorized by the Board.

RESPONSE OF AEC REGULATORY STAFF TO MEMORANDUM OF LAW OF CITIZENS COMMITTEE FOR THE PROTECTION OF THE ENVIRONMENT

In its Memorandum of Law, the Citizens Committee raised many issues which are in support of its Findings of Fact and Conclusions of Law relative to Applicant's motion for a 50% power testing license. As indicated above in our response to the Citizens Committee's proposed findings, the regulatory staff took issue with those matters which we deemed to require specific rebuttal, although the Citizens Committee paper was replete with generalities and statements not based on the record of this proceeding. We will therefore not repeat our corrections of intervenors' contentions, but will instead direct our attention to the contention that "the regulatory staff's review of the Application for Indian Point, Unit No. 2, was inadequate."

At the very outset, we must make it clear that the issues to be decided in the subject proceeding are those issues which the Commission enumerated in the Notice of Hearing dated November 17, 1970. Absent from the list of issues is the question of adequacy of the regulatory staff review of subject applications. The Board is to make safety findings in this, as in any contested proceeding. It is the function of the regulatory staff as a party in such a proceeding (10 CFR 2.701b) to place in the record its review of the application.

The statement by intervenors that the regulatory staff is a "proponent of a particular nuclear power reactor" is wholly without merit. At the very outset of this proceeding, staff counsel, in an opening statement, outlined the searching and intensive review prior to our safety evaluation which the regulatory staff gave to subject application, and the many amendments to said application which were offered by the applicant as a result of such staff review. In such opening statement, staff counsel advised the Board, parties, and public that the applicant has the burden of proof with respect to its application, and the staff is in reality supporting the staff review. The intervenor would shift that burden, which is clearly on the applicant in the Commission's Rules and Regulations (10 CFR 2.732), and pass it to the staff, or the applicant and staff.

With respect to the specific items of alleged "inadequacy" of staff review, our responses are as follows:

1. The regulatory staff did indeed confer with and advise the State of New York with respect to its emergency plans for Indian Point, Unit 2. The testimony adduced at the hearing and our response above to this line of contention clearly demonstrates that fact. Intervenor states that "no changes or modifications were recommended by the regulatory staff in the emergency plans submitted by the applicant or the State. Again the Citizens Committee either does not understand the regulatory review which takes place prior to any hearing, or finds it convenient to forget that such review took place. We did review the emergency plans, and we are satisfied that they meet the requirements for such plans.

2. The list of documents which the regulatory staff gave to the Citizens Committee on August 25, 1971, relating to ECCS review was an accurate list at that time.

3. The regulatory staff position on official notice of the myriad of documents, periodicals, etc., which intervenor has requested has been clearly enumerated in the three briefs on the subject which the staff has filed in this case.

4. The contention that the regulatory staff "will permit the need for power from Indian Point to compromise the health and safety of the public" is without merit and not based upon the record of this proceeding. Throughout the course of subject hearing and during the review preceding the hearing, the regulatory staff has concentrated its activities to assure that the plant will in no way pose a threat to the health and safety of the public. The staff evaluated Indian Point Unit No. 2 on the basis of standards set by the Commission, and did not attempt to evaluate this plant in the context of "the safest possible plant." The health and safety of the public is paramount in the staff review, and every item of regulatory activity, from the safety evaluation to the technical specifications to the operating license, evidences such concern.

5. We have indicated, in response to Citizens Committee finding 10.(b.), that the Compliance Division of the regulatory staff must provide findings on the adequacy of the restoration work caused by the fire before a license can be issued.

6. In our view the Citizens Committee's contentions with respect to the review conducted by staff on the 50% testing application are without merit. The review was conducted in accordance with Section D.2. of Appendix D to 10 CFR Part 50. The balancing that was done was in accord with the above section, and the intervenor's implication that the balancing detracted from the staff's commitment to the health and safety of the public is not substantiated by the record of this proceeding.

We find the Citizens Committee Proposed Findings to be without merit, and urge the Board to:

1. Make appropriate findings on the issues specified in 10 CFR 50.57(c) for operation of the Indian Point Unit No. 2 facility at 50% power level for testing purposes.
2. Balance the factors for 20% power level for testing in accordance with 10 CFR Part 50 Appendix D Section D-2.
3. Authorize the Director of Regulation to issue an amendment to Operating License No. DPR-26 authorizing operation up to 20% of power level for testing purposes.
4. Certify to the Commission, without recommendation, the record in this proceeding relating to the 50% testing application for its balancing of factors under Appendix D, and determination on the remaining 30% power level for testing.

5. Upon specific approval of the Commission, authorize the Director of Regulation to issue an amendment to Operating License No. DPR-26 authorizing operation up to 50% power level for testing purposes.

Respectfully submitted,

Myron Karman

Myron Karman
Counsel for AEC
Regulatory Staff

Dated at Bethesda, Maryland
This 10th Day of March, 1972

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "Response of the AEC Regulatory Staff to Proposed Findings of Fact of Citizens Committee for the Protection of the Environment," dated March 10, 1972, in the captioned matter, have been served on the following by deposit in the United States mail, first class or airmail, this 10th day of March, 1972:

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