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BERLIN, ROISMAN AND KESSLER
1910 N STREET, NORTHWEST
WASHINGTON, D. C. 20036

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March 9, 1971

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U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

Myron Karman, Esq.
Office of General Counsel
U.S. Atomic Energy Commission
Washington, D.C. 20545
Mail Station: P 506A

Dear ~~Mr. Karman~~ Myron:

Enclosed herewith are the Round Two Questions submitted on behalf of the Citizens' Committee for the Protection of the Environment, and designated Set I.

Sincerely,

Tony Roisman

Anthony Z. Roisman

Enclosure

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March 9, 1971

Lex Larson, Esq.
Leboeuf, Lamb, Leiby and
MacRae
1821 Jefferson Place, N.W.
Washington, D.C.

Dear Lex,

Enclosed herewith are the Round Two Questions submitted on behalf of the Citizens' Committee for the Protection of the Environment, and designated Set H.

Sincerely,

Anthony Z. Roisman

Enclosure

cc: Samuel W. Jensch, Esq.
J.D. Bond, Esq.
Dr. John C. Geyer
Arvin E. Upton, Esq.
J. Bruce MacDonald, Esq.
Honorable Louis J. Lefkowitz
Angus McBeth, Esq.
Stanley T. Robinson, Jr.
Myron Karman, Esq.

AZR/mb

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Instructions
for Questions, Sets H and I

1. The request to answer in detail or explain or justify should be interpreted to mean provide more than mere reference to source and summaries of conclusions. Actual reproduction of critical portions of the source should be provided and the bases for conclusions should be explained. While FSAR references are helpful, they are not normally source but merely summary. The request for detail should be interpreted in light of the practical objective of avoiding lengthy cross-examination. The more data produced now, the less will have to be elicited in cross-examination.

2. Where qualitative terms such as "possible", "probable", "credible", "indications", "conservative", etc. are used please define the term in some meaningful manner so that your underlying assumptions are understood.

3. Do not apply the narrowest interpretation to what is sought. For instance, in questions D-41 and D-46 reference was made to "faults" in the reactor vessel. The answers referred to compliance with codes. No answer was given to the questions as related to code approved defects which are apparently called "indications". Questions H-6 et seq. try again for the same data. Remember intervenors are not necessarily cognizant with all the technical lingo.

4. When reference is made to an accident and no specific accident is mentioned, please use the accident which would produce the worst (i.e. most conservative) conditions for purposes of the question.

5. With respect to each answer or part thereof, please identify the individual who adopts the answer or part as his or her testimony and identify all documents or references upon which he or she relies for the answer. At the end of all of the answers please have each person who has answered sign an oath of affirmation.

Questions and Requests for Documents Submitted to Con Ed by the Citizens Committee for the Protection of the Environment (March 9, 1971)

1. In answer C-1 you use the term probability. Define this term as it is used in the answer. Is there any possibility of an explosive rupture of an element of the primary loops?

2. Explain why a double ended pipe break in the hot leg could not involve a rupture in which pieces of metal from the pipe could be propelled against the inside of the containment as a result of the rapid release of pressurized water from the loop.

3. If the answer to question H-1 is yes, what would be the force in psig with which the largest, the median and the smallest piece (specify size) would strike the containment. In this answer use conservative values at least with respect to the following elements:

   a. age of the pipe
   b. location of the rupture at a welded joint
   c. proximity of the pipe to the containment wall.

4. If the answer to question H-1 is yes, provide the following information:

   a. How many individual fragments would result from this rupture?

   b. Have you analyzed the force of these fragments (in psig) and if so what is that force?

   c. Have you analyzed the probable route of fragments and if so, how many will come in contact with other equipment or pipes within the containment?

   d. With respect to c., have you analyzed the effect of these fragments on the objects they could strike and the result of that collision on the ability of the post-accident function of equipment or pipes within the containment? If so, please provide the analysis in detail.

In this question also use conservative values for the factors specified in question H-3.

5. Answer question H-4 (regardless of the answer to H-1) with respect to the water released by the rupture and also with respect to the broken ends of the ruptured pipe assuming they remain attached to the remainder of the pipe.

6. Were the steel plates used in the reactor vessel and the welds for the vessel subjected to ultrasonic testing? Radiograph or X-Ray testing? With respect to all such tests of the plates and the welds provide the following information (Please do not merely refer to the information provided in pages Q4.1.1-1 to Q4.12-1 of the FSAR):

a. At what stage(s) of the manufacturing (including ingot stage) and installation of the plates and the manufacturing and installation of the vessel were the tests conducted and by whom?

b. How much of each plate was tested with the instrumentation perpendicular to the plate and how much was tested with oblique (shear wave angle beam) shots (note that Tech. Spec. p. 3, 1-5 suggests that only certain plates received 100% testing of both perpendicular and oblique beams)?

c. When were the tests conducted?

d. Were flaws (regardless of whether they were within Code specifications) of any size permitted in the plates and if so, what was the largest size permitted for each kind of plate or weld used in the reactor vessel?

e. Were maps made of the flaws and can their exact location be shown on the reactor vessel as it is now installed? If so, please provide the map.

f. How many flaws and of what sizes exist in the reactor vessel plates and welds?

g. Define the term "indications" in answer D-41.

7. Will ultrasonic testing of plates or welds which are perpendicular to the plate or weld detect all or any cracks that are parallel to the beam of the equipment? If the answer is yes, please explain in detail.

8. Will radiograph or X-ray testing which is perpendicular to the plate or weld detect a vertical crack if it is less than 2% of the thickness? If the answer is yes, please explain in detail.

9. If the welds are tested by radiograph or X-ray, what standards are used for approving the weld? For instance, in the 1968 Section 3, ASME Boiler and Pressure Vessel Code for Nuclear Vessels, pages 172-178, it explains that visual comparison of the picture is made with the gauge charts (pp. 174-178) and the gauges show what size and how many flaws can remain. Were these or similar gauges used and if so how many of which size holes in the weld were permitted?

10. Technical Specifications 4.2 set forth the pre-operational and in-service structural surveillance of the reactor vessel and primary system boundary. With respect to this specification, please answer the following questions (References are to the Tech. Spec.):

a. Will baseline data come exclusively from ultrasonic, visual and surface (please describe) techniques conducted after the reactor vessel is installed? What will be done with the data from earlier tests (see answer to H-6) and will there be any radiograph or X-ray testing for baseline data? (4.2(a))

b. Define the term "defects" and explain the role of the AEC in evaluating and investigating these defects. (4.2(c))

c. Describe in detail every difference between the inspection Code referred to in 4.2.3 of the Tech. Specs. and the Code referred to at Q. 4.1.1-1 by the AEC. Attach a copy of each Code.

d. Describein detail the basis for the claim that ultrasonic testing is an acceptable alternative for radiographic examination. In particular, what kind of flaws (defects, indications, etc.) will be detected by radiograph which cannot be detected or cannot be detected as well by ultrasonic and if there are none justify your conclusion. (4.2.3(b))

e. At 4.2-3 of the Tech. Specs. and elsewhere in 4.2 (see 4.2-12; Table 4.2-1; and Notes (1)(4.2.-17)) you indicate that radiation levels in the reactor vessel, among other factors, present special problems which prevent certain in-service inspections until new equipment is developed. With respect to this, answer the following:

1) Explain the meaning of answers A-11 and A-24.

2) Describe in detail the present level of development of these testing techniques (and the techniques themselves) including who is now developing them, how far along has development come (design, prototype, full tests of equipment, etc.), any firm commitments that you have on delivery date of these techniques, how design and manufacture procedures have been prepared for these developments, anticipated cost of the new techniques.

3) Justify in detail the delay in in-service testing referred to in Items 1.1, 1.2, 1.3 and 1.7 and what is the outer limit of that delay?

f. Justify the delay in inspection referred to in Item 1.4.

g. Explain in detail how the visual examination referred to in Items 1.5 and 1.6 will be able to detect any internal growth in flaws (defects, indications, porosity) in the welds.

h. Provide a copy of the Code Section referred to in Item 1.15.

i. Justify your refusal to conduct tests referred to in the first paragraph of Item 4.2 both in terms of the impossibility of conducting the test and your belief that such tests are unnecessary.

11. Explain in detail the manner in which the following factors taken together and separately can affect the growth of flaws (indications, defects) in the reactor vessel including its welds and the primary piping system (if there is no effect, justify the conclusion; if there is, in your opinion, an insignificant effect, justify the conclusion regarding the extent of the effect and the insignificance of the effect.)

a. Long term (10, 20, 30 years) exposure to the 550-650 degree temperatures of the primary coolant;

b. Long term (10, 20, 30 years) exposure to the radio-activity in the primary coolant - Supply a copy of the report referred to in Answer C-5;

c. The impact of emergency core cooling water on the reactor internal and external walls in the case of double ended cold leg break. For this answer provide also an analysis using the formulae in D-50 (In Answer D-50 (page 2) to what does "stainless steel cladding" refer)as well as the following formula from Reference 2 to that answer:

$$\text{sigma theta} = E\left(\frac{A \cdot \Delta T}{2}\right)(1 - nu)$$

E = Young's modules of elasticity
A = coefficient of thermal expansion
ΔT = temperature
nu = poisons ratio

In this case provide the following analysis:

1) temperature of the interior of the reactor walls for each second following the break;
2) level of the water in the reactor for each second following the break (or confirmation of the relevance of FSAR Figures 14.3.2-1 and 14.3.2-5);
3) temperature of the emergency cooling water (both accumulator and the main supply)at the earliest possible moment of contact with any uncovered (with water) portion of the reactor wall and time at which contact will be made;
4) total stress on the reactor wall at the point of contact as well as analysis of the total effect (in terms of pressure created) within the reactor of the cooling water contacting the reactor walls (assume the contact occurs at a point on a plate where the maximum permissible flaw (defect, indication) exists for a reactor in operation for 25 years - make the same assumption for contact with a weld;
5) all other relevant factors which will demonstrate the maximum possible stress at the weakest, possible point; and
6) answer the question with respect to the simultaneous impact of cooling water on the exterior of the reactor vessel as a result of the pipe break and the containment spray.

12. Justify the substantial time lag between the examination of the irradition samples and relatively few samples used for purposes of adequately keeping track of the shift in NDTT. See Tech. Specs. 3.1-6 and 4.2-16. Explain in particular, inter alia how the samples will adequately detect the pressence of unusually high radiation leakage from a specific area of the reactor near a specific section of the vessel wall. Also explain the manner in which answer to 14.3.1-1 is relevant to this. Why does that question mention 8 samples and the Tech. Specs. (4.2-16) refer to 6 samples?

13. With respect to the answer to Question 4.8 and the reactor vessel stress analysis explain in detail whether

a. the calculations were made with respect to the particular reactor vessel involved in Indian Point No. 2 or only with respect to that type of vessel. If the latter justify this decision.

b. the calculations take account of the presence of flaws (defects, indications) in the vessel and their growth as a result of the factors discussed in H-11. If not, justify the validity of the analysis and the answer.

c. how the fact that actual shift in NDTT has to await periodic examinatin of test samples (Tech. Spec. 3.1-6) affects the validity of the analysis and the answer.

14. Discuss in detail the data which supports the conclusions which comprise the answer to Question 4.9.1. On the basis of the answer it will be determined whether a request will be made to see WCAP 7332.

15. The answer to Question 4.10 indicates that Class I plant components are designed to the ASME Code prior to 1968. In ONRL-NSIC-21 (Technolofy of Steel Pressure Vessels for Water-Cooled Nuclear Reactors) the following comments apppear with respect to these ASME Codes:

1. p. 150

The maximum temperature at which light water reactor pressure vessels are designed to operate is 650 degrees Fahrenheit. No problems attributable solely to the loss of tensils properties due to temperature are anticipated for the materials being used in the construction of nuclear pressure vessels provided the steels possess at least the minimum tensible properties stated in Table N-424, Section 3 of the ASME Code. Adherence to these properties can be assured by imposing supplementary requirements on the materials suppliers such as those given in 5-7, high temperature tension tests of ASTM spec 8533.

At least one pressure vessel customer currently requires that tensile data be obtained at 550 and 650 degrees Fahrenheit for the shell plate material as part of the fabrication test program. (Emphasis added)

2. p. 51

Another area of concern is the relevancy of present requirements of authorized inspectors, as established by the National Board of Boiler and Pressure Vessel Inspectors, with regard to nuclear pressure vessels. The existing requirements are heavily weighted toward the needs of Sections 1 and 8, rather than 3. Consequently, presently qualified inspectors may not have sufficient understanding of the design requirements and non-destructive test methods required for nuclear vessels. We therefore recommend the upgrading of qualifications of code inspectors of nuclear pressure vessels to a level of competency achieved by professionally educated and experienced personnel. (Emphasis added)

3. p. 410

In order to assure that an adequate stress analysis of the vessel has been made, the Section 3 rules stipulate that a stress report be prepared, certified by a professional engineer and filed with the proper authorities at the point of installation. The rules also provide that experimental stress analysis methods, either strain gauge or photoelastic, may be used to verify specific design areas, when theoretical calculations are unavailable, or for determining fatigue reduction factors for cyclic operation. The results of such tests are to be included in the design report. The code specifies only that a complete set of stress analysis calculations shall be made and reported. It deos not specify that the calculational methods used must yield correct or conservative results as verified by experimental data, or that such evidence shall be offered in support of the calculations. The code does require that the stress report be certified by a registered, professional engineer experienced in pressure vessel testing. The Code does not specifically say that the professional engineer must be experienced or qualified in stress analysis. The inspector who affixes the code stamp is specifically not responsible for the completeness or correctness of the design calculations as set forth in this stress report. [emphasis added]

With respect to the underlined material indicate whether the additional requirement suggested has been applied to Indian Point No. 2. If so, how and if not, why not.

16. Do the ASME Codes have different requirements today than the ones used and referred to in the answer to Question 4.10? What about the draft ASME Codes or the AEC Reactor Development and Technology program standards dated July 31, 1970? To the extent that any of these are more stringent than the Code used for the Class I Components explain in detail the difference and why the more stringent requirement is not needed for greater safety. If the answer requires more than you are prepared to provide at this time then give the answer only with respect to the reactor vessel.

17. Describe in detail the tests of pipe line vibration for pipes penetrating the containment which will be conducted after plant operation begins. Give inter alia, frequence of tests, extent of piping tested, and what kind of corrective measures will be taken.

18. Justify the failure to consider jet forces and tornado loads in the design of the large openings of the containment.

19. With respect to the answers to Questions 5.14(a), 5.14(b), 5.14(d), 5.14(e), 5.15 please provide copies of the relevant pages of the Indian Point No. 3 PSAR.

20. Justify the reliability of the equipment hatch during design basis accident and earthquake loads when the liner shows deformations which can be tested only for pressure (tensile stresses) and not for accident loadings (compressive stresses). See Question 5.14c-1. Explain how ductile behavior under tensile stress can adequately represent ductile behavior under compressive stress.

21. Explain in detail the operation changes with respect to the reactor when Indian Point No. 2 is connected to the Con Ed load frequency control system. When will this occur? Indicate to what extent the control of the reactor power level will be determined automatically by load demands from Con Ed's customers and the effect on the reactor power output of a sudden drop in power demand or a sudden increase in power demand on the system. Explain how these variations in nuclear power output of the reactor will affect the various safety features of the plant.

22. In answer B-19 you indicate that pressure in excess of 5 psig will not affect the function of the redundant flame recombiner unit. On FSAR, Question 6.8(a)-2 you state that the unit is designed to operate in pressures of 0-5 psig and indicate that it will not

be operated until pressure reaches that level. See also pages Q6-8(b)4 and 5-2 and Q 6.8(b)4 and 5-3.
If pressure is in excess of 5 psig up to 40 psig and if the amount of hydrogen in the containment atmosphere exceeded 2% could the recombiner unit be used at that time. Explain fully a yes answer in light of the design of the unit. If the answer is no, what system would be used?

23. The recombiner unit uses containment air to cool its exhaust which is allegedly below 300°F. Discuss the impact of the heat addition to the containment caused by the recombiner unit in the context of double-ended hot leg and cold-leg pipe ruptures. In particular how will operation of the recombiner affect the predicted post-accident pressure level in the containment and how will this affect the conservative estimates of radioactive leakage to the atmospher and the control room.

24. Describe the situation in which oxygen will be added to the containment atmosphere for operation of the recombiner unit discussing when (in terms of hours after the worst accident) the oxygen will be needed and the method for injecting this oxygen in to the containment. At the time when oxygen concentration is less than 12% what will be the likely chemical composition of the containment air, its temperature, its pressure and its moisture level.

25. Will use of the recombiner units require a decision to be made within the control room or will the units be started automatically when required. Specify the highest level of hydrogen which will be permitted to accumulate before the units are put in use and how many hours after the accident this will occur.

26. Explain in detail the nature of the uncertainty associated with the catalylic recombination system for hydrogen removal. See Question 6.10-1. In particular does this uncertainty stem from uncertainty regarding the composition of the post-accident containment air or is it only uncertainty regarding operation of the catalytic recombiner itself under reasonably predictable conditions.

27. Assuming 3/4 of the on-site spent fuel storage capacity is filled and assuming Indian Point No. 1 and 2 have been operating at full power level for 300 days, how much plutonium will be present at the Indian Point site in the:

a. Spent fuel storage

b. Reactor core for each reactor separately

As a basis of comparison relate this to the amount of plutonium released (best estimate) in fallout from the above-ground explosion of the largest plutonium nuclear weapon of the United States.

28. Justify the answer given to question H-10(e) in light of Criterion 45 in ORNC-NSIC-24 (p. 107).

29. On page 3 of Answer E-17 and D-1 you indicate that because Indian Point No. 2 is not in the "high density accident area" associated with glidepaths for take-offs and landings in the immediate vicinity of the airfields no analysis needs to be done of the possible crash of a 300,000 lb. aircraft into the reactor. Justify this decision and discuss or reveal, <u>inter alia</u>, the following factors:

a. Show flight routes and holding patterns for all three major New York airportsas well as the Westchester County airport for all routes and holding patterns within a 10 mile horizontal distance from Indian Point. If you are unwilling to answer because you believe 10 miles is too large explain in detail your reasons and answer the question for the acceptable distance.

b. Indicate with respect to these routes the average number of airplanes on the route each year and their average altitude.

c. Indicate the number of mid-air collisions between airplanes one of which will land or has taken off from the airports involved,in the last ten years.

d. Indicate what data was obtained from which FAA officials with respect to your conclusion that the crash of an airplane into the reactor is so incredible that no analysis of the effect of that accident is required.

30. Describe in detail how the security measures referred to in Answer A-58 and the answer to FSAR Q12.6 would prevent sabotuers such as those who have recently bombed the U.S. Capitol and other buildings around the country from entry to the security area

a. by tuneling under the security fence;

b. by cutting the security fence;

c. by using light weight ladders or pole vaulting over the fence; or

d. by entering the water discharge or inlet pipes and cutting through whatever screening exists there.

31. Further discuss the available protection from shaped charges fired from a boat on the river, a low flying aircraft or a truck. With respect to this question indicate which structures of the Indian Point plants would be damaged and in what manner by the maximum sized shape charge fired from a bazooko, a mortar and a rifle mounted grenade launcher as well as the largest charge which can be dropped from helicopters or aircraft available for rental in the area. This analysis should include analysis of damage to pipes, wiring, towers and other similar structures.

32. If any radioactivity is released off-site as the result of a design basis accident, describe in detail the steps which private citizens living within five miles of the plant could take to reduce their exposure to this radioactivity to the lowest practicable level.

33. To what extent have you provided or will you provide information to these citizens of the proper use of these exposure limiting techniques.

34. Do you have any plans to alert citizens of off-site radioactivity levels in excess of normal operating releases (not necessarily exceeding 10 CFR, part 20 levels) and if so what is this plan? If you do not have such a plan who does and what have you learned about the effectiveness of that plan for giving early warnings to citizens of these releases?

35. How soon after a design basis accident would the public notification referred to on pages 14-15 of the Radiation Contingency Plan be made. What are the criteria to be applied by the coordinator in judging the severity of the situation and deciding to give the

notification. What requirements are imposed upon the Con Ed individuals so notified with respect to the specific actions which they must take and the time schedule required for such actions.

36. To the extent that you do not have plans or do not know the details, of state or federal plans to educate the general public as soon as possible on the steps to be used to reduce exposure to any abnormal releases of radioactivity (whether below 10 CFR Part 20 limits or not) from the Indian Point plants and to the extent you do not have plans or do not know the details of state or federal plans to inform the public immediately when an abnormal radioactive release occurs present a justification for these failures. In the course of this discussion explain the basis for failing to advise state and federal authorities at once of any abnormal release of radioactivity. See pages 12-14 of Radiation Contingency Plan.

37. By what method are the recirculation sump screens and containment sump screens prevented from becoming clogged with the materials which they are designed to screen out. Describe the quantity of anticipated debris and compare to the area of the screens involved.

38. In the design basis LOCA describe the containment humidity, pressure, heat and hydrogen content and the fuel clad temperature under the following conditions for the first 100 seconds after the double-ended pipe break:

a. failure of the ECCS (See Answer A-9 and ORNL-NSIC-24 (pp. 68-69).

b. failure of the out of containment safety injection system to provide any water and operation of only 3 of the 4 accumulators.

39. With respect to the charcoal filters used for iodine removal in the post accident environment please set forth the effectiveness of the filters for removal of iodine during the first 100 seconds and during the remainder of the first day following the design basis LOCA with specific reference to the containment humidity and its effect on the filter efficiency as discussed in the answer to Q14.10. To what extent were these ORNL test statistics (FSAR 14.3.5-6 used in calculating the iodine removal capacity of Indian Point No. 2 as stated in the FSAR. Justify the validity of the predicted organic iodine removal rate in light of the lack of full scale testing referred to in the last paragraph of FSAR 14.3.5-5.

40. What specific systems not considered in TID14844 operate to make impossible or not credible for Indian Point No. 2 the conceivable conditions referred to in paragraph 1 on page 17 of TID14844. Do not explain in detail how the systems work but do explain in detail how the conservative values obtained in analyzing those systems relate to the specific kinds of incidents which could occur and produce the results considered in paragraph 1. In short relate the safety systems to the causes of the TID14844 conditions and demonstrate how much of those conditions are eliminated using conservative values for the functions of the safety systems on Indian Point No. 2.

41. Has Con Ed performed a failure tree and an ARRM reliability analysis model comparable to the one done on the Dresden plant and illustrated in HN-190 (ARRM) p. 1-51? If so, provide two copies and indicate the probability of failure for Indian Point No. 2 in light of the analysis. If not, justify this failure.

42. Discuss the alleged adequacy of the effectiveness tests on the containment spray system in light of the differences between the Applicant and the staff for the spray iodine reduction factor and the difference with respect to the amount of plateout. Relate this discussion to the comments by Board member Pigford in the Initial Decision on Indian Point No. 3.

43. If no more than 3% of the fuel melted in a LOCA would there be any possibility of a steam explosion that could rupture the vessel. Discuss in detail the analysis used for your answer including the probability assigned to a 3% fuel melt down.

44. Describe in detail the effect in the reactor vessel from the emergency cooling water coming in contact with the fuel rods and the general release of energy and steam pressure within the reactor vessel. For this answer assume the worst LOCA (double-ended break, cold leg) and consider the following factors as well as all other relevant factors

a. variations of fuel rod heat in different parts of the reactor both vertically and horizontally.

b. the effect of clad swelling and clad bursting in light of Table 3.8 (p. 56) of ORNL-NSIC-24 (Emergency Core-Cooling Systems for Light-Water-Cooled Power Reactors and the Discussion contained therein (pp. 59, 69, 70-75, 86, 92)) the discussion on p. 267-268 of Fundamental Nuclear Energy Research (1969) a Supplemental Report to the Annual Report to Congress, and the extent to which tests have been conducted with clusters of fuel rods with design basis internal pressures.

c. the existence of a metal water reaction with the use of 2100°F. as the temperature at which metal-water reaction produces energy at a rate comparable to the decay heat (ORNL-NSIC-24 (p. 50, 55-58)), the use of temperatures shown in FSAR Figure 14.3.2.-23 and the predicted reflooding rate shown on figures FSAR 14.3.2-1 and 14.3.2.-5 in light of the statement in the second paragraph on p. 85 of the ORNL-NSIC-24.

d. the reliability of the estimates on how quickly emergency cooling water from accumulators and from the safety injection system reach the reactor including consideration of back pressure created in the reactor vessel, delay in the operation of valves in the post accident environment, the untestable exist nce of short circuits in ECCS motors (ORNL-NSIC-24 (p. 62) and the relatively high unreliability of diesel backup power systems (ORNL-NSIC-24 (pp. 62-63))) and delay in diesel start-up (Answer to B-22).

e. the actual delay involved in covering the entire core as the result of the factors discussed in FSAR 14.3.1-18 (first paragraph) and the reason that steam pressure will not flow out the down comer before sufficient head can be built up in the downcomer.

f. the percentages of clad burst shown on FSAR 14.3.1-20.

g. consideration of whether the tests referred to in the fifth paragraph of FSAR 14.3.1-21 were conducted with fuel rods with design basis internal pressures and justification for the conclusions stated in the second paragraph of FSAR 14.3.1-22.

h. the generation of pressure data referred to in Fundamental Nuclear Energy Research (1969) pages 268-269.

i. the pressure of some fuel rods enriched at a higher level than others.

j. a justification for the assumption of any adiabatic conditions at the clad surface.

45. Does the design leak rate from the containment apply only for the first minute after an accident? If so, please explain the basis for this. If not, please explain the statement at the top of page 14.3.5-14.

46. Explain the procedure for removing operators from the control room and at what time this will be done following an accident as referred to at FSAR Q14.16-4.

47. For what reason were the particular assumptions regarding retained fission products in the core used in FSAR Q14.8-3(C.1.)? Aren't these inconsistent with AEC assumptions? Explain.

48. Provide the analysis in Q14.8-4 for the first 10 days following the design basis LOCA.

49. Provide two copies of the test reports referred to in the answers to Q14.3.3 and Q14.3.5. If these are proprietary documents provide a detailed summary from which we can assess the need for obtaining the proprietory document and from which we can obtain as much information as possible.

50 Explain in detail the basis for the assumption that accident discussed in Q14.6-2 will result in the radioactivity being initially released under water. What if the dropped fuel assembly were perforated by contact with some object above the water. Explain the significance of the Westinghouse analysis when it is conducted in water which does not contain the many radioactive elements which would be present in the accident situation.

51. Which tests conducted with reference to Q6.3 were conducted in a solution containing the combination of all elements in the appropriate ratios present in the containment liquid following an accident. Justify the validity of any tests not so conducted.

52. What procedures are used to determine if there is any mercury in water which will be in the containment after an accident and how is all of the mercury removed from the water to meet the requirement of paragraph 4.1 of FSAR Q6.3-13.

53. Justify the use of test temperatures for aluminum corrsion below post accident temperatures in the containment, FSAR Q6.3-19 and 20. Explain the effect of the aluminum corrosion on the equipment which has aluminum in the containment. FSAR Q6.3-9.

54. Justify the conclusion that Nordel used in the tank valves will not be adversely affected by exposure to sodium hydroxide solution on the basis of a six-month exposure test (FSAR Q6.4-1) in light of the length of time specified between tests of the valves as shown in Tech Spec. 4.5 (I.B.) (4.5-3).

55. Discuss your conclusion to disregard the possibility of a failure of the reactor vessel in the design criteria for Indian Point No. 2 in light of the ACRS statement quoted in the AEC answer to A-44 (letter dated January 11, 1971).

56. Major meltdown is not a postulated accident for this plant (see answers to questions 8, 9 and D-69).

a. Can it be inferred from this that there is 100% certainty on the Applicant's part that the ECCS will function satisfactorily in any "credible" loss of coolant accident?

b. If the answer to a. is affirmative, can the Applicant justify his faith in the ECCS without periodic functional testing of the entire system under accident conditions?

c. Is such testing contemplated and does it include flooding the reactor core with borated water from the accumulator tanks under accident conditions of temperature, pressure and humidity?

d. Does the AEC Staff believe there is 100% certainty that the ECCS will perform satisfactorily in any "credible" loss of coolant accident and, therefore, that the probability of major meltdown is zero?

e. What assumptions, either explicit or implicit, are made in the FSAR question Q14.1-1 (which is concerned with the iodine reduction factor of the air cleaning systems necessary to meet the 10 CFR 100 guideline values) as to the effectiveness of the ECCS?

57. Do any of the test reports relied upon in the FSAR represent reports which have excluded unfavorable test results even if the unfavorable test result was presumably irrevelant. If the answer is yes, identify the reports and justify your reliance upon them. If you do not know the answer justify your reliance on the test reports.

Questions and Requests for Documents
Submitted to AEC by the Citizens Committee
for the Protection of the Environment (March 9, 1971)

1. Describe the difference between the ability to determine the result of a LOCA from the LOFT program and from the programs now being used for analysis. Concentrate in particular on the weaknesses or shortcomings of the present testing which justify the substantial expense entailed in developing a working model. Also discuss the answer in light of the following statements contained in pages 1363-1369 of AEC Authorizing Legislation, Fiscal Year 1971, Part 3, March 11, 1970:

> ". . . (1) LOFT is the focal point which provides a fundamental sense of direction to water reactor safety investigations, (2) as a live reactor in an accident mode <u>it makes the investigators face reality</u>, and (3) if provides a central vehicle to build and hold a competent technical staff in a vital national program" (Emphasis added)
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> * * * *
>
> These tests, in which electrically heated assemblies simulating full-size reactor fuel pins are cooled by sprays and flooding, are needed to increase confidence in emergency cooling system performance under design and off-design conditions.
>
> * * * *
>
> To date, the tests performed indicate that emergency core cooling systems, as designed, will perform their intended function over a wide range of cooling and temperature conditions. However, this confidence level must be extended to the higher operating power densities characteristic of future nuclear plants. In addition, it is desirable to simulate more closely the reactor system conditions predicted during possible emergencies, and to extend the temperature range to higher levels to resolve performance limits.
>
> * * * *
>
> On the basis of single rod tests it is apparent that the ballooning (swelling) of fuel rods during a loss-of-coolant accident potentially can cause appreciable coolant channel reduction. Out-of-pile multirod experiments have been initiated at ORNL to examine the effect of rod interference and

randomness of failure on such blockages. Initial results indicate that at least 40 percent of the original coolant channel will remain. While it is believed that this will allow passage of sufficient emergency core cooling water, based on these tests and on tests in the Full Length Emergency Cooling Heat Transfer Test (FLECHT) Program, described, previously, this is still subject to further experimental test work at ORNL.

*   *   *   *

Initial results indicate that large numbers of channels can be blocked without substantially affecting ability to cool the fuel bundle in an emergency situation. However, the extent of blockage which could occur in a bundle, has not yet been fully explored, but is a part of the presently planned program.

2. Fundamental Nuclear Energy Research (1969) a Supplemental Report to the USAEC Annual Report to Congress for 1969 (pp. 265-269) refers to several areas in which further research is required to understand the LOCA. The report states:

> Specific information on several topics must be available if the consequences and potential hazards which may result from a loss-of-coolant (LOC) incident in a boiling or pressurized water reactor are to be realistically analyzed. The physical and mechanical properties of the reactor core materials must be defined at temperatures above their melting points. The behavior of these materials when exposed to steam at high temperatures must also be determined. In addition, all of these data are needed in assessing the adequacy of the design of the emergency core cooling system. Specific projects under the nuclear safety research program are providing the basic data.

Please explain in substantial detail how it is now possible to definitively determine that a large reactor can be safely placed near a high population area when the LOCA has not been "realistically analyzed" in all aspects. Please identify which specific aspects of a LOCA at Indian Point No. 2, could be said to have not as yet been "realistically analyzed". Justify your statement.

3. Identify the number, names and qualifications of personnel from the AEC who have had responsibility for Indian Point No. 2. State their specific duties and responsibilities and approximate number of days spent performing these duties. List their additional responsibilities during the same period with respect to other reactors.

4. Provide two copies of the latest Report of the Advisory Task Force on Power Reactor Emergency Cooling.

5. How does the answer F-59 indicate compliance with the AEC requirement that radioactive releases be kept as low as practicable?

6. In light of the stated purposes of witnessing tests (Answer A-30b) justify the adequacy of your mere spot checking of actual testing as a means of performing your safety analysis. Discuss whether the checks are made with knowledge in advance by Con Ed and the method used for deciding how many tests to check and how frequently the checks are to be made.

7. Provide copies of the operating progress reports on the LOFT program and the LOFT semiscale test, etc. referred to on FSAR 14.3.1-14.

8. To what extent have the uncertain ties and concerns expressed in the ACRS letter of February 26, 1968 on the Report of Advisory Task Force on Power Reactor Emergency Cooling been satisfied with tests conducted since that time and to what extent is further testing required including the LOFT program.

9. Answer each question asked of the Applicant in light of your detailed analysis of the FSAR. For any questions which you cannot answer on the basis of your independent analysis justify the thoroughness of your investigation of that aspect of the plant.