CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

Indian Point Station, Unit No. 2

Docket No. 50-247

Applicant's Responses to Round Two Questions
Submitted by The Citizens' Committee for the
Protection of the Environment (Set H)
on March 9, 1971

Part II

March 31, 1971

Question No. H-29

Question: 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, inter alia, the following factors:

a. Show flight routes and holding patterns for all three major New York airports as 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 inclusion that the crash of an airplane into the reactor is so incredible that no analysis of the effect of that accident is required.

Answer: a. The flight routes within 10 miles of the Indian Point Plant are shown on Figures 29-1*, 29-2*, and 29-3*. The Brewster holding pattern shown on Figure 29-4** serves Westchester Airport, which is used primarily by smaller aircraft (BAC-111 and FH-227 type aircraft). The New York Metroplex is shown in Figure 29-5.

* These are portions of the New York Center VOR Peak Day Charts, September 5, 1969, obtained from the New York Center for air traffic control, on which Applicant has located Indian Point and a 10 mile radius circle.

** This is a portion of the New York Sectional Aeronautical Chart, 2nd Edition, effective January 7, 1971 issued by the U.S. Department of Commerce, on which Applicant has located the Monroe and Brewster holding patterns, Indian Point, and a 10 mile radius circle.

b. The number of aircraft which pass within 10 miles of the Indian Point Plant on the designated airway routes and on direct flight paths between intersections are tabulated below. The numbers tabulated are for September 5, 1969, which was the day during 1969 when the New York Center for air traffic control handled the most flights.

Route	17,000 feet and below	18,000 feet and above
J-37	--	11
V-126	98	--
V-292	42	--
Direct	158	3
	298	14

c. The records indicate that one mid-air collision of the kind described has occured within the last ten years. On December 4, 1965, an Eastern and a TWA airliner collided near Carmel, N. Y. The TWA airliner landed safely at JFK and the Eastern plane crashed in Connecticut.

d. The above information was obtained by applicant from the records of the New York Center for air traffic control.

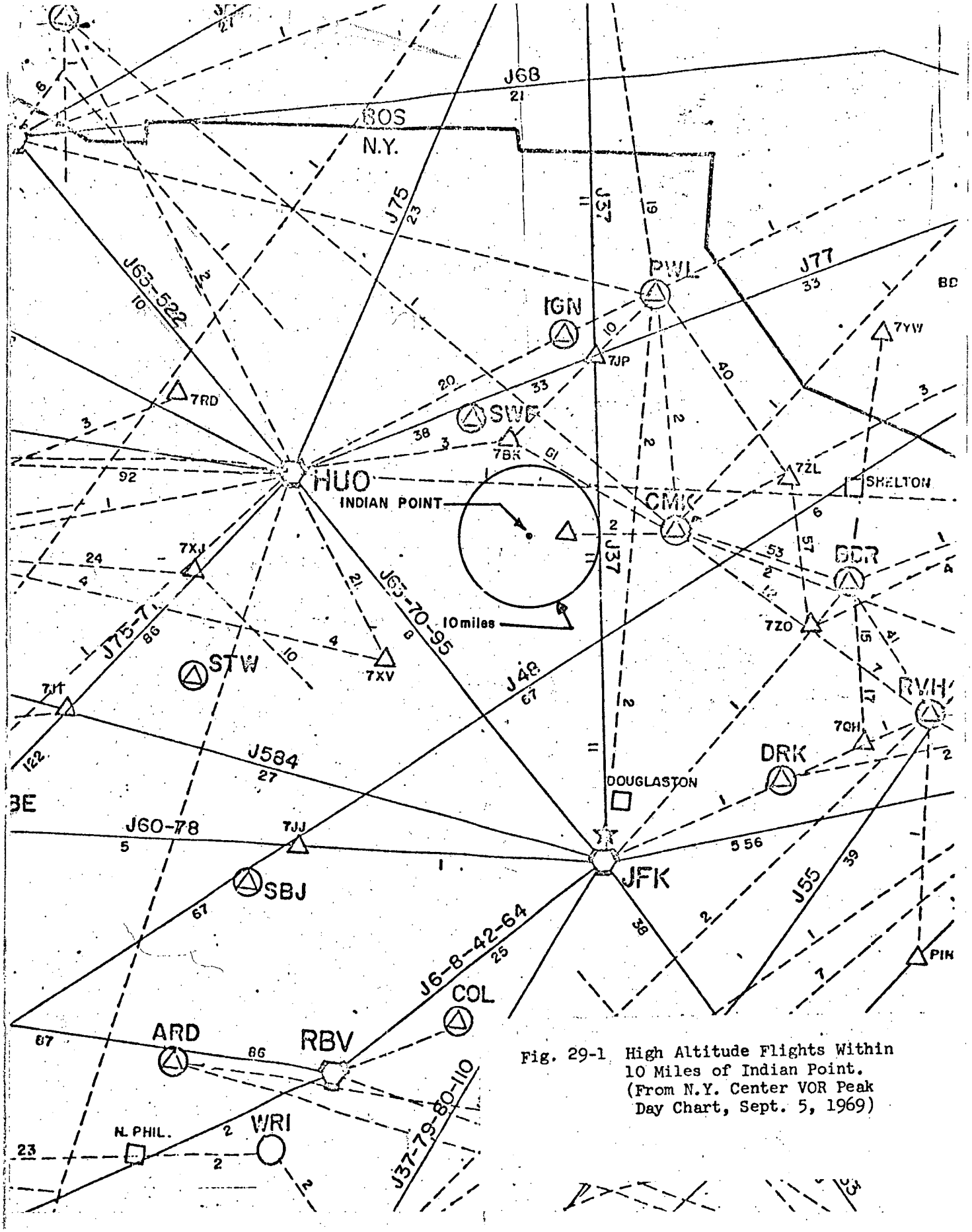

Fig. 29-1 High Altitude Flights Within 10 Miles of Indian Point. (From N.Y. Center VOR Peak Day Chart, Sept. 5, 1969)

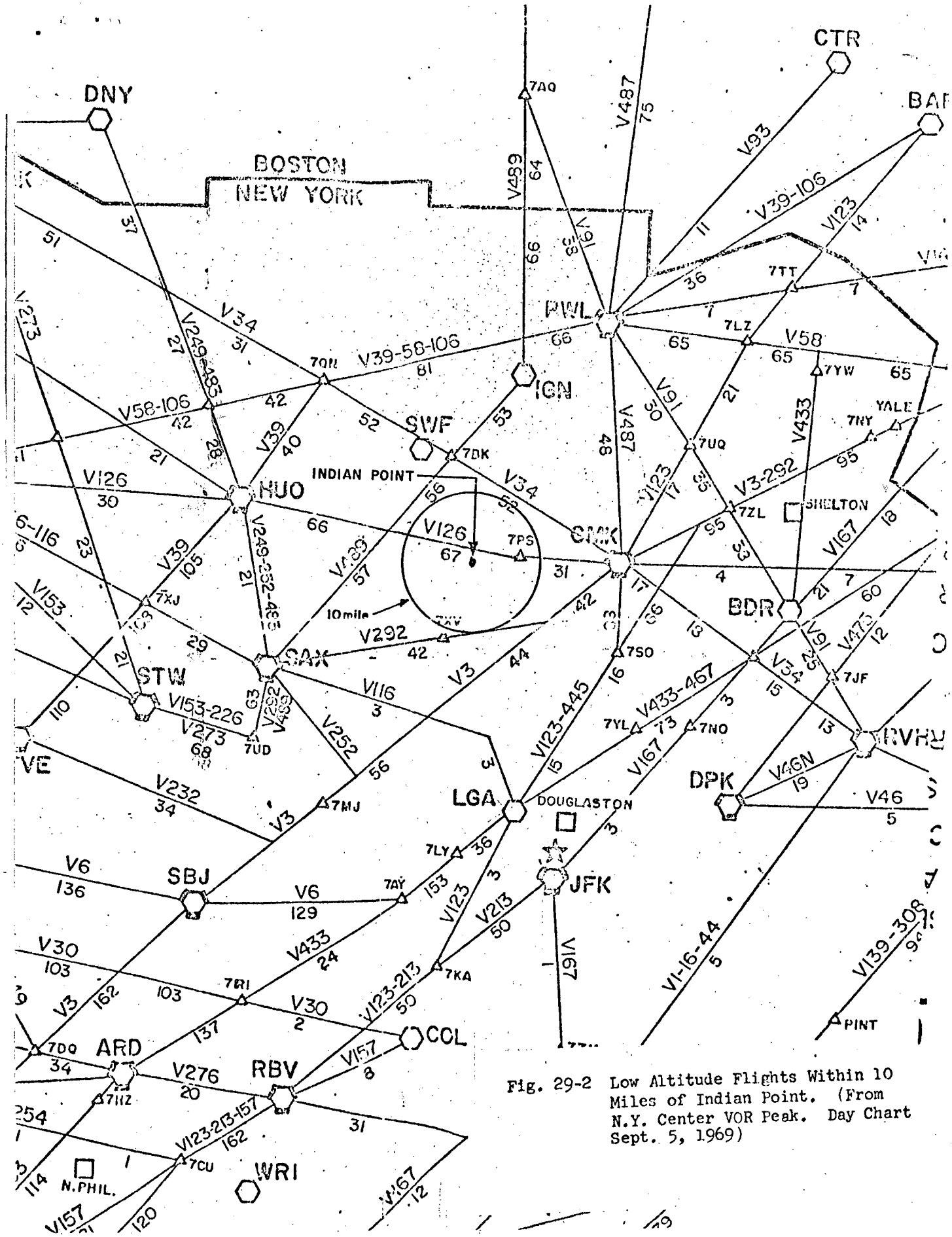

Fig. 29-2 Low Altitude Flights Within 10 Miles of Indian Point. (From N.Y. Center VOR Peak. Day Chart Sept. 5, 1969)

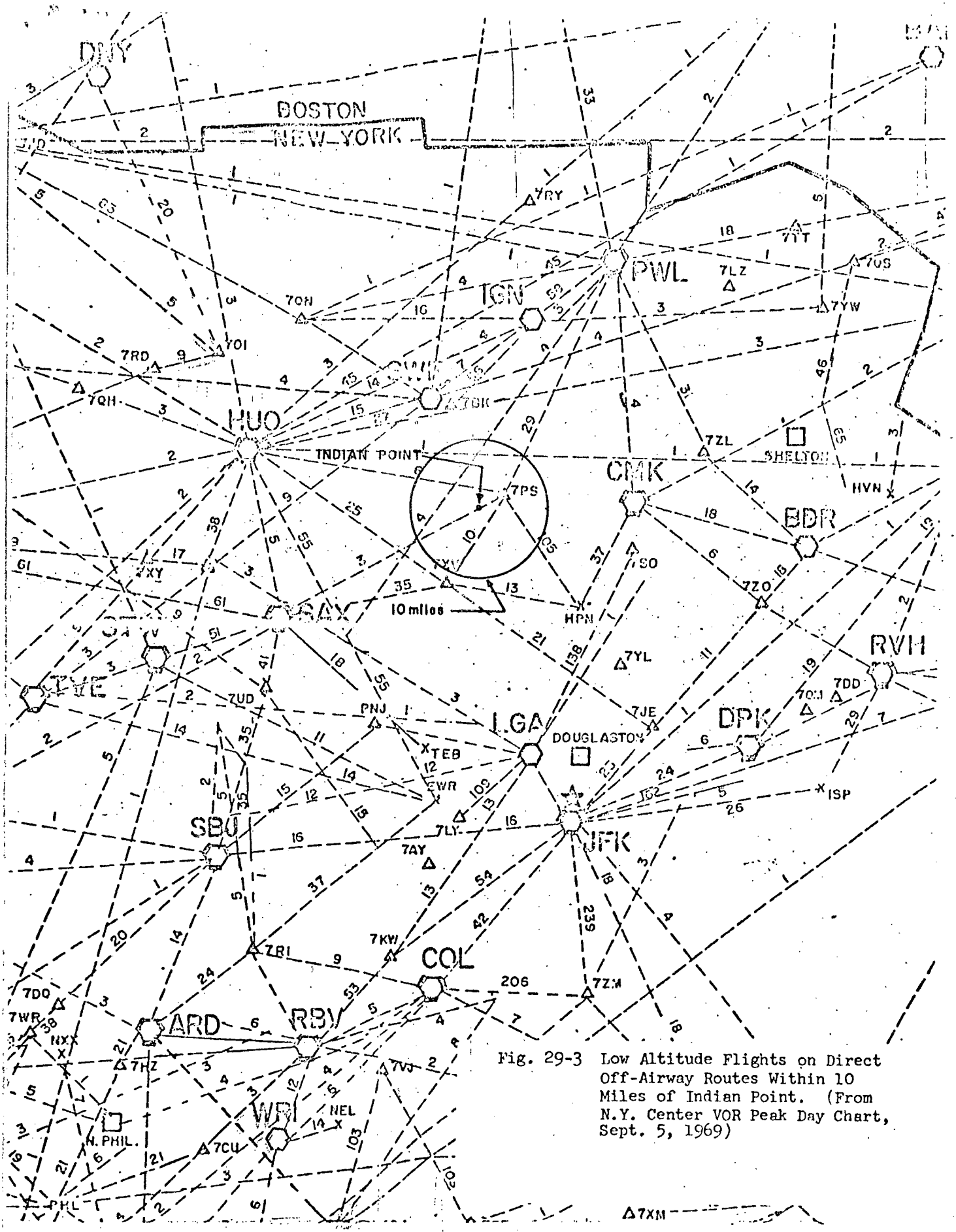

Fig. 29-3 Low Altitude Flights on Direct Off-Airway Routes Within 10 Miles of Indian Point. (From N.Y. Center VOR Peak Day Chart, Sept. 5, 1969)

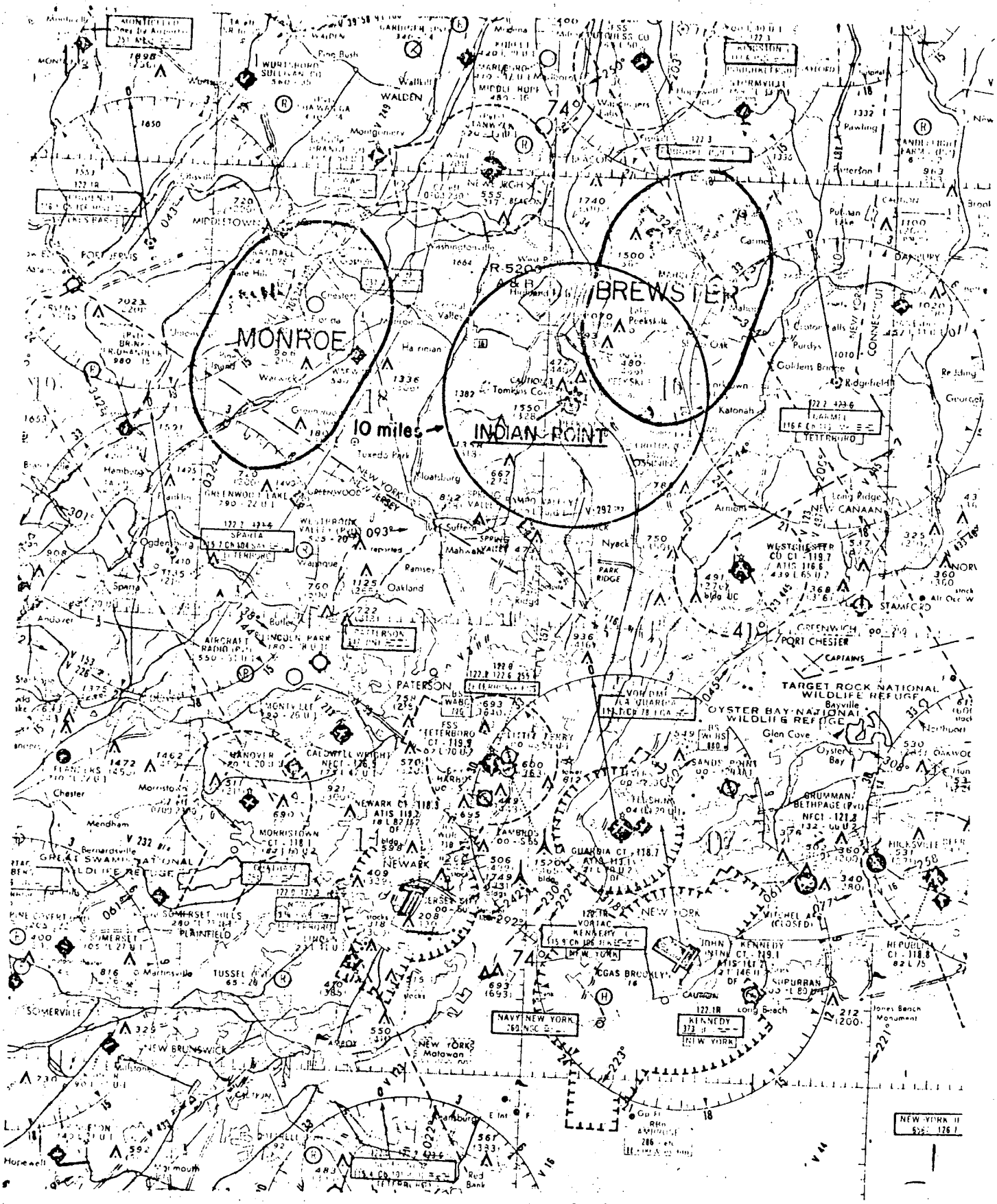

Fig. 29-4 Aircraft Holding Patterns Near Indian Point.

NEW YORK METROPLEX

(PART OF FAA AIR TRAFFIC CONTROL SYSTEM – APRIL, 1970)

As illustrated by Coast and Geodetic Survey, this graphic shows: (1) Delay absorbing holding pattern airspace areas. (2) Arrival clearance limit holding pattern airspace areas. (3) Major arrival routes into the above areas. (4) Terminal arrival and departure routes for a southwest operation, for aircraft arriving and departing Newark, La Guardia and Kennedy airports and associated satellite airports. Arrival clearance limits and routes into the holding pattern airspace remain unchanged when other runway combinations are used. Changes in terminal arrival and departure routes can be expected during these periods.

All possible routes are not shown. Routes other than shown or other deviations from this system may be made at the direction and discretion of air traffic control.

CAUTION
This illustration is not to be used for navigational purposes.

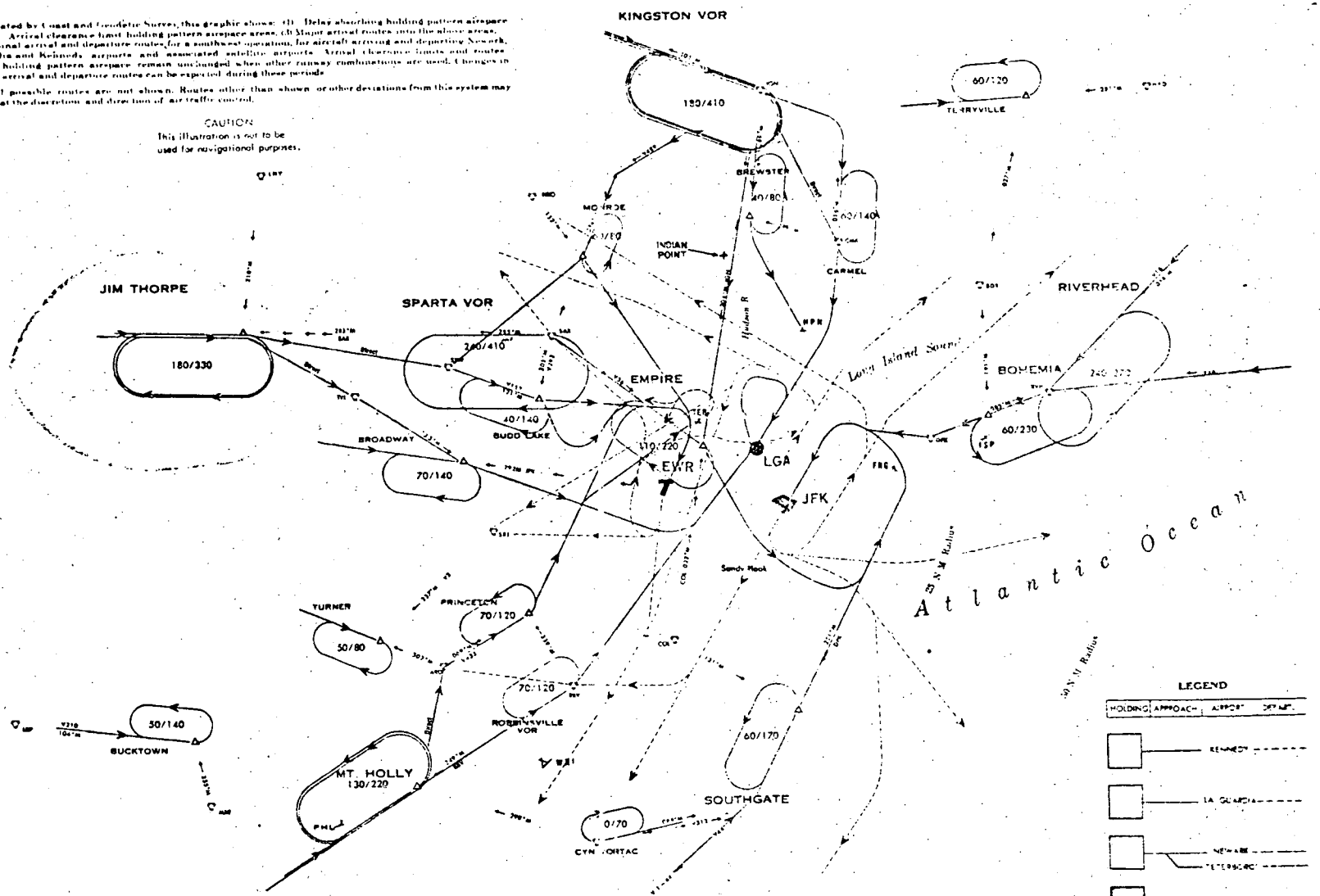

Figure 29-5. New York Metroplex (Holding Patterns)

Question No. H-30

Question: 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.

Answer: The guards will make routine patrols of the plant site including the properties within the site boundary. Any evidence of tunneling or cutting of the fence will be immediately reported to the Central Control Room. In addition, the general watch foreman, watch foreman or assistant superintendent inspect the palnt security perimeter once per watch checking that all outer doors and windows are properly secured. All doors and entries to the nuclear area will be equipped with a system of locks and alarms with indicators for these alarms in the plant security room. Procedures and means exist for obtaining outside assistance if needed. Sabotage of the conventional parts of the plant would not cause any radiation hazard to the public.

Question No. H-31

Question: 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.

Answer: Applicant has not performed an analysis to determine the consequences of hostile acts such as those described in the question. The plant does have several features which afford protection from some such acts. These include a site which is patrolled by security forces to control access, a reinforced concrete containment (4 feet-6 inch walls and 3 feet-6 inch dome) surrounding the reactor and the reactor coolant system, and redundancy and diversity in protection systems which are designed to withstand single failures no matter what the cause of such failure. These features serve to prevent or limit the release of radioactivity as a result of some such acts.

Question No. E-32

Question: 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.

Answer: In the event of a design basis accident, Applicant will act according to the radiation contingency plan, as described in the FSAR Question 12.5, and in the answer to Questions No. 6 and 14 of the Atomic Safety and Licensing Board at the January 19, 1971 hearing. The "New York State Emergency Plan for Major Radiation Accidents Involving Nuclear Facilities," dated February 1971 (Applicant's Exhibit 2) describes actions to be considered to minimize public exposure to radioiodine in the event it were to be released off site as the result of an accident at Indian Point 2. The New York State Health Department would determine whether to initiate such protective actions, in accordance with that Plan.

Question No. H-33

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

Answer: Applicant has not provided information to these citizens concerning use of exposure limiting techniques in the event of an accident at Indian Point Unit No. 2. It is Applicant's view that it would not serve a useful purpose to do so because the actions to be taken at the time of an accident would be dependent upon a wide variety of circumstances. In accordance with the State's emergency plan (Applicant's Exhibit 2), if an accident occurred, the affected population would be notified at that time as to actions to be taken to minimize radiation exposure if the New York State Health Department determined that such notification were necessary.

Question No. II-34

Question: 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?

Answer: Releases of radioactivity from the plant, including unusual releases, will be reported to the U.S. Atomic Energy Commission in accordance with the provisions of 10CFR Part 20 and Sections 3.9.A.3 and 6.6 of the Technical Specifications (Supplement 1 to Staff Exhibit No. 1). The first of these sections provides: "A report shall be submitted to the Commission at the end of each 6-months' period of operation as required under Specification 6.6.4. If quantities of radioactive material released during the reporting period are unusual for normal reactor operations, including expected operational occurrences, the report shall cover this specifically...." Copies of reports on releases are available in the public document room of the Atomic Energy Commission in Washington and are furnished to New York State officials as provided in FSAR Question 11.1 and 12.5.

In the event of accidental releases, Applicant would alert the proper Federal, state and local authorities in accordance with the radiation contingency plan described in FSAR Question 12.5. The affected public would be promptly notified of such releases by the New York State Health Department if it determined that protective actions were called for.

Question No. H-35

Question: 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.

Answer: Please refer to the answers to Questions No. 6 and 14 of the Atomic Safety and Licensing Board at the January 19, 1971 hearing.

In the event of a design basis accident, the New York State Health Department would be notified immediately through the New York State Civil Defense Commission Warning Point. The Health Department would determine whether protective actions were called for and would notify the affected public promptly by the most expeditious means available. Factual information on a design basis accident would be reported to the public by the Health Department and Applicant as it became available.

The radiation contingency plan sets forth the specific actions to be taken by Con Edison personnel in the event of site contingencies or general contingencies. The time schedule required for such action would depend upon the particular duties of the individuals involved and the nature of the accident.

Question No. H-36

Question: 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.

Answer: See the foregoing answers to Questions H-32, H-33, H-34 and H-35.

Question No. H-37

Question: 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.

Answer: In the Indian Point #2 plant, two separate sumps are provided, either of which can be used in post-accident core cooling; one from which the internal recirculation pumps take their suction, and a second which connects to the external recirculation loop.

The type of debris which has been considered to be possible, though unlikely, to fall to the floor of the containment during the hypothetical loss-of-coolant accident is as follows:

1. Metal objects dislodged by the coolant jet, such as position switch housings, instrument cases, valve handles, etc.

2. Pieces of wire and insulation therefrom

3. Particles (flakes) of paint

4. Grit eroded from concrete

5. Pieces of thermal insulation (metallic or mineral)

6. Glass from broken light fixtures

7. Corrosion products dislodged during accident

8. Casual materials such as dirt, scraps of paper, etc.

Most of the debris listed above is more dense than the spilled coolant and can be expected to fall to the containment floor and remain in the immediate area of the reactor coolant loops. Transport across the floor of the containment is extremely unlikely since the average velocity of the water flow across the floor to the sump is very low (of the order of 10 ft. per min.) due to the large floor area and unobstructed flow path to the sumps.

Some of the debris may be lighter than water and will tend to float on the surface of the fluid on the floor. Since the sump is below the floor and the depth of water over the sump entrance is about 3 feet, this floating debris should not be carried down into the sump inlet.

That debris which is of approximately the same density as the flowing water or which has a geometry conducive to transport with the slowly flowing stream will be carried to the sump inlet. Scraps of paper, particles of paint, corrosion products and pieces of thermal insulation are the types of debris expected to be carried to the sump inlet plus some of the erosion products of small particle size (sand) from the concrete.

Each of the two sumps is provided with multiple barriers against the particulate transport as follows:

a. A coarse floor grating through which the flow passes vertically downward at a velocity of less than 1 ft./sec.

b. A settling chamber containing baffles which cause flow to change direction.

c. A medium mesh screen through which flow passes generally upward at less than 1 ft./sec into the pump suction compartment.

Debris which is smaller in size than the openings in the medium mesh screen can be easily passed through the recirculation pumps.

The design guards against clogging by providing for gravitational separation of nearly all of the solid material (by flotation or settling) in slow-flow areas ahead of the screens, and with ample screen area to accommodate any material which, because of near neutral buoyancy, might be carried to the screens.

The areas of the coarse grating and screen are each 48 sq. ft. in the internal recirculation sump, and 15 sq. ft. in the external recirculation sump. The quantity of material available to be transported to these screens, for reasons cited above, would be ineffective in blocking flow.

Question No. H-38

Question: 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.

Answer: The safety analysis presented in the FSAR, Section 14.3.4 encompasses Part (a) of this question, because it assumes a failure of any element of the emergency core cooling system (ECCS) at the time of the design basis accident.

If we were to infer from the questioner's reference to NSIC-24, pp. 68-69 that total (that is, multiple) failure of ECCS is to be hypothesized for the first 100 seconds, the following conditions would be obtained:

Containment humidity would be 100% (i.e., saturated at the pressure obtained).

Containment heat (temperature) and pressure would be essentially the same or slightly lower than those presented in the design

basis case (FSAR Appendix C, Figure II-2.3A and Figure 14.3.4-2).

Fuel cladding temperature would reach the melting temperature (3375°F) at the hottest spot about 45 seconds after the accident; at 100 seconds, about 9% of the cladding would have melted.

Because of the Zirconium water reaction which would accompany exposure of the cladding to steam at this condition, a quantity of hydrogen equal to 0.2% by volume of the containment would be produced.

In response to Part (b), the hypothetical failure of the out-of-containment portion of the emergency core cooling systems is postulated, following delivery of the contents of three accumulators, the following conditions would be obtained:

Containment humidity would again be 100%.

Containment temperature and pressure would more nearly correspond to the design basis case than in the Part (a) instance.

Fuel cladding temperature would reach about 2200°F at the hottest spot. A minor amount of Zirconium-water reaction (3.2% of the cladding at the hot spot) would occur; however, the core-wide extent of this reaction would involve a negligible fraction of the clad, and the resulting hydrogen production would be insignificant.

Question No. H-39

Question: 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.

Answer: Charcoal filters are used for organic iodine removal during the post accident period. During the first 100 seconds and for the time period following, a single-pass effectiveness of 70 percent is used for design. This effectiveness is conservatively below the results obtained by ORNL and reported in ORNL-TM-2728, which showed 88 percent single-pass effectiveness at the design conditions of 100 percent relative humidity and 270°F. The tests were run at ORNL specifically to determine the effect of the design basis conditions which are assumed to exist during the first 100 seconds as well as the first day following the accident.

The validity of the organic iodine removal rate (single pass effectiveness), is justified by the tests performed. The quantity of organic iodine present has no effect on the single-pass effectiveness.

Question No. II-40

Question: 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.

Answer: The Emergency Core Cooling System limits the release of radioactivity from the core in the event of a loss of coolant accident by assuring that fuel temperatures will be well below fuel melting temperatures. The Weld Channel and Penetration Pressurization System and the Containment Isolation Valve Seal Water System seal leakage paths from the containment to substantially eliminate leakage and thereby offset effects of atmospheric diffusion conditions suggested.

The Containment Spray System and the Containment Air Recirculation and Filtration System cool the containment atmosphere to assure containment integrity.

Question No. II-41

Question: 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.

Answer: A failure tree and reliability analysis comparable to the one done on the Dresden plant and illustrated in HN-190 (ARRM) p. 1-51 has been performed. The probability of failures for Indian Point Unit No. 2 in light of the analysis is contained in this document. A copy of this analysis will be furnished separately.

Question No. H-42

Question: 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.

Answer: 1. Iodine Removal Effectiveness of Containment Spray System

The iodine removal effectiveness of the containment spray system was calculated in the Indian Point Unit No. 2 FSAR. This calculation is based on a simplified model of the absorption of iodine into the spray drops. Some of the assumptions made in this simplified analysis were questioned as to their effect on the calculated results by Dr. Pigford in the Initial Decision on the Indian Point Unit No. 3 construction permit. These assumptions include the neglect of the mass transfer resistance in the liquid phase (liquid film resistance), the effects of the drop size spectrum produced by the nozzles, and coalescence of drops.

Westinghouse has carried out a R&D and testing program to investigate these phenomena, and their effect on the iodine removal effectiveness of the containment spray system. The following results were obtained from this program:

a) The question of the liquid film resistance was resolved by a more rigorous model for the calculation of the uptake of iodine into the drop. This model is based on molecular diffusion into the drop as the only mass transfer mechanism and, therefore, describes the maximum possible resistance in the liquid phase.

b) The effects of coalescence were described in a detailed analysis, in which the number of collisions between the drops are calculated, and each collision is assumed to result in a coalescence (i.e., the formation of a single larger drop). The results of this analysis showed that the reduction in the mass transfer surface area due to coalescence is approximately 10% for a drop fall height of 100 ft.

c) The complete drop size spectrum produced by the spray nozzles was used in the new model. This drop size distribution had been obtained from measurements made with a SPRACO 1713 nozzle, which is of the same design as the SPRACO 1713A nozzle used in the Indian Point Unit 2 and 3 containments.

Since the analysis of the iodine removal effectiveness is based on the spray drop size distribution obtained from these nozzles, a testing program was conducted to confirm the performance of this nozzle. This testing program included pressure and flow measurements, patternization and spray angle checks, and drop size measurements. The results of these tests showed that the 1713A nozzles meet or exceed all performance requirements. Specifically, the drop size spectra obtained from the nozzle show that the actual mean drop diameter is significantly smaller than the drop size used in the calculations. Since the iodine removal effectiveness of the spray increases with decreasing drop size, this result adds an additional margin of conservatism to the calculations.

The uncertainties about the calculation of the iodine removal effectiveness, therefore, have been removed by this comprehensive program of analytical and experimental investigation of the mathematical models and physical parameters involved.

2. Plateout Model

The guidelines given in the Technical Information Document, TID-14844, are used by the DRL staff to calculate the amount of iodine released to the containment atmosphere.

A similar, but slightly more conservative model was used for the dose calculations presented in the Indian Point Unit No. 2 FSAR.

The doses calculated with the iodine removal effectiveness demonstrated by the above-described comprehensive program are below the 10CFR100 guidelines using either plateout model.

Question No. H-43

Question: 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.

Answer: The plant is designed to preclude such a meltdown. The emergency core cooling system for Indian Point Unit No. 2 is designed to maintain both cladding and fuel temperatures well below melting for rupture sizes up to and including the double-ended severence of the largest reactor coolant pipe. In fact, the fuel temperature never approaches its initial full-power value at any time in the course of the loss-of-coolant accident.

Question No. 44

Question: 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 existence of short circuits in ECCS motors (ORNL-NSIC-24 (p. 62) and the relatively high unreliability of diesel backup power systems (ONRL-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.

Answer: The heat transfer capabilities of pressurized water reactor (PWR) emergency core cooling systems under simulated loss-of-coolant accident (LOCA) conditions has been thoroughly investigated in the PWR Full Length Emergency Core Heat Transfer (FLECHT) tests. In these tests, the axial and radial power distributions employed correspond to those that would be found in actual PWR fuel assemblies for the design code. Furthermore, the effect of variation of fuel generation on the heat transfer coefficient during

reflooding was analyzed by varying the peak power from a maximum of 1.4 kw/ft to a minimum of 0.69 kw/ft. The test results indicated that initially the heat transfer coefficients are the same for different heat generation rates, while at later times the lowest heat generation rate shows the highest heat transfer coefficient.

In the LOCA analyses the axial and radial power distribution is properly included in analyses of the temperature transient for the peak power fuel rod. In addition, the reflooding heat transfer coefficients used in these analyses are conservatively based on the results of the FLECHT test performed with a peak power of 1.24 kw/ft, which is higher than the power generated by the peak pellet in the Indian Point Unit No. 2 core at the time of reflooding.

During a loss-of-coolant accident, the clad temperature may get sufficiently high so that bursting or swelling of the clad would occur by virtue of the internal gas pressure and the significant reduction of clad strength. To evaluate the core geometry deformation resulting

from such an event, Westinghouse performed the Single Rod Burst Test (SRBT) and the Multi Rod Burst Test (MRBT) programs. The SRBT provided the deformation characteristic of the Westinghouse fuel rod cladding for a wide variety of conditions including the design basis internal pressure of 2250 psia to simulate the range of variables covered in the LOCA analyses. Based on these results, the combination of parameters which yielded the maximum deformation were determined and used in the MRBT to experimentally obtain the worst core geometry distortion during heatup following a LOCA.

The MRBT basis test configuration was an 8x8 array of 3-foot-long rods, surrounded by a heated shield. The center 4x4 rods were pressurized at pressures including the design basis internal pressure of 2250 psia; the remaining rods were not. All of the rods were brought to normal operating temperature by external heaters on the shield. Then, an electrical resistance power transient was imposed on all the rods to rapidly increase temperature until the pressurized rods ruptured. The temperature

distribution in the array during the transient and the sequence of ruptures was monitored. The 4x4 was removed and examined by mounting and sectioning to provide flow blockage maps at several elevations.

Results of the MRBT show that the burst locations are staggered axially along the fuel rods and that, to some degree, rod to rod contact does occur. However, the remaining flow area is always sufficient to ensure adequate core cooling. Analytical evaluations for a typical double-ended cold leg break, considering flow redistribution due to the geometry distortion and rod-to-rod contact, have shown that the peak clad temperature increases less than 100°F above the peak temperature without geometry distortion. This analysis was performed for 100% of the rods burst in a fuel assembly and the worst case blockage. This yields a higher peak clad temperature than the blockage associated with the total bursts presented in FSAR pg. 14.3.1-20.

Furthermore, the effect of severe flow blockage on heat transfer effectiveness during reflooding was studied in the FLECHT Tests. The test results

indicate that due to atomazation of the entrained water and to the rapid flow redistribution to the bottom flooding heat transfer effectiveness is not impaired by the geometry distortion which may derive from bursting and swelling of the fuel cladding during a LOCA.

The possibility of fuel rod embrittlement due to chemical reaction between Zircaloy cladding and steam has also been considered in the SRBT program. New and previously burst tubes were raised to high temperatures (2100 to 2700°F) and after varying lengths of time at this temperature the samples were quenched in a containment of water. The results indicated that clad tubing exhibiting metal-water reactions of as much as 16% maintains its integrity.

As explained in the first paragraph of the FSAR 14.3.1-8, the heat generated by the reaction of the Zircaloy clad with steam is considered in evaluating the fuel rod temperature transient in addition to the fuel rod decay power. The reaction heat generation is conservatively calculated using the Baker's parabolic reaction rate even when no

steam flow is assumed to be available for cooling the core (such as the time period between the end of blowdown and the beginning of entrainment).

The loss-of-coolant accident analyses presented in the FSAR were revised to include the improved calculational techniques developed for the evaluation of the LOCA. The results of the new analysis indicate a maximum peak clad temperature of 2015°F for the double ended cold leg break.

The reflooding calculations have been modified to incorporate the results of the FLECHT tests. The basic assumptions used in this new core reflooding model are:

1. For the first few seconds of reflooding, the flooding rate is independent of the loop resistance (FLECHT).

2. The downcomer head required to drive the steam generated in the core is calculated with the loop resistance evaluated with the homogeneous K-factor. The result is modified by an experimental multiplication factor to account for the departure from the homogeneity.

3. The flow through the core and the quench front velocity as a function of the inlet core flow rate is obtained from experimental results (FLECHT).

4. No credit is taken for the entrained liquid droplets falling back to the core.

The core reflooding rate determined by this method for Indian Point Unit No. 2 is reduced from a flooding rate of approximately 9 in/sec to 2 in/sec at the time entrainment begins. Subsequently, due to the increase of the water level in the downcomer, the core flooding rate increases to about 2.5 in/sec.

The evaluation of the double-ended cold leg break temperature transient using the above flooding rate indicated a peak clad temperature of 2015°F as previously calculated. In this analysis, the fuel rod clad temperature after reflooding decreased less rapidly than indicated by Figure 4 in Appendix 14B of the FSAR. The total amount of Zr-H_2O reaction at the hot spot was 2.8%.

It should be pointed out that a peak clad temperature of 2015°F and a maximum local Zr-H_2O of 2.8% are well below the LOCA limits as indicated by Figure 5 in Appendix 14B of the FSAR.

The loss-of-coolant analysis presented in Chapter 14.3.2 of the FSAR considers the existence of the prevailing back pressure in calculating both the accumulator and the safety injection system flow rates. The accumulators are activated when the system pressure drops below 600 psig. Since the accumulators are a passive system which requires only the opening of check valves, no malfunction delays in delivering accumulator water to the vessel are expected.

With the exception of the internal recirculation coolant pumps which are not used during the initial injection period, all the safety injection pumps are outside the containment. Furthermore, it should be noted that the residual heat removal pumps which are located outside the containment provide a redundant system to the internal recirculation pumps.

To assure operability of motor-operated valves and pumps in accident environment, Westinghouse has performed tests in post-LOCA environment. A description of these tests and their results is in answer to AEC Question 7.8 of the FSAR. The results of these tests indicate that motor-operated valves and pumps will be operable in a post LOCA environment, as well as under normal conditions. It follows that no additional delay in the operation of the valves will be experienced under LOCA conditions beyond that required to start the diesel. Furthermore, the operability of motor-operated valves and pumps is periodically tested so that potential short circuits in the motor will be detected.

The problem discussed in the FSAR, Page 14.3.1-18, first paragraph refers to the beginning of the reflooding phase when the ECCS water reaches the bottom of the core. At this time, even though unlikely, the loop seals may be filled with water. Should this happen, the steam generated in the core would have no escape paths and would increase the

pressure in the system. It is clear that since the downcomer height (16.4 ft) far exceeds that of the loop seals (8.5 ft), the pressure buildup in the system will blow the liquid out of the loop seals much before being able to push the downcomer water through the break. As stated in the FSAR, the resulting delay in recovering the bottom of the core would be insignificant.

The tests described in the fifth paragraph of FSAR 14.3.1-21 were conducted with unpressurized rods. It should be noted that even if the rods were pressurized, the pressure would have caused the rod to burst, thus relieving the rod internal pressure, much before reaching the peak clad temperature of 2800°F. Subsequently, quench tests were performed with various combinations of internal pressures from 100 to 2250 psi and clad temperatures from 1000 to 1900°F (approximate maximum temperature obtained for rods which do not burst). No loss of integrity occurred in any specimen. Quench tests from high temperatures (2200 to 2500°F) were also performed with previously burst rods. No shattering was observed when up to 16% of the clad thickness was oxidized.

As indicated previously, the combined effects of clad bursting and swelling were considered in determining the effects of the worst geometry distortion that can be experienced during a LOCA in a PWR core. The calculated increase in peak clad temperature was less than 100°F.

An essential condition for the generation of a pressure surge referred to in the above reference is the presence of molten material. As indicated by the results of loss-of-coolant accident analysis, the maximum peak clad temperature (2015°F) is well below the melting temperature of the Zircaloy-4. In addition, the fuel temperature is well below the melting point. It follows that no pressure surge due to molten material-water contact will occur during a LOCA.

Variations in enrichment is considered along with other uncertainties in evaluating fission gas pressure in fuel rods. Considering these variations, the open volume in the fuel rod is designed to limit the internal pressure to less than 2250 psia.

The assumption of adiabatic conditions at the clad surface is conservative since this yields the maximum fuel rod temperature rise. In effect, when the clad surface is adiabatic as heat is removed from the rod, all the energy generated goes to increase the fuel rod temperature. It should be pointed out that even during these periods of time when no steam is assumed to be available for cooling the core, the Zr-H_2O heat of reaction is evaluated using the parabolic rate equation.

Question No. 45

Question: 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.

Answer: The Indian Point Unit No. 2 FSAR contains many accident analyses with many different assumptions. The statement at the top of Page 14.3.5-14

> "...the containment leaks at its design rate for one minute at which time leakage terminates."

is taken from the description of an analysis entitled <u>One Minute Isolation of Containment-Gap Release</u>, which begins on Page 14.3.5-13. For this particular analysis, a leak rate assumption is based on the proper functioning of the isolation valve seal water system and the penetration pressurization system to block leak paths. This is the most realistic case in the unlikely event of a LOCA.

Other analyses have been performed in which the containment leak rate is assumed at its design value for the duration of the accident, and the resultant doses are within 10CFR100 guidelines.

Question No. H-46

Question: 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.

Answer: In FSAR Q14.16-4, the doses indicated are based on a continuous thirty-day exposure. As a result, the control room operators will not receive the dose stated in FSAR Q14.16-4, during their normal working hours following a design basis accident. The operators may wear protective clothing and full face respirators to further reduce radiation exposure in the control room or when going to and from the control room.

Question No. II-47

Question: 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.

Answer: As stated in FSAR Q14.8-3, Part C-1, hydrogen generation by core radiolysis is calculated assuming that 50% of the halogens, 100% of the noble gases, and 99% of all other fission products are retained in the core. This is the same as the TID-14844 release model except for treatment of the noble gases. With the TID-14844 release model, noble gases are released from the core to the containment atmosphere, and thus do not contribute to hydrogen generation. The basis stated in FSAR Q14.8-3 (C-1) retains the noble gases in the core where they contribute to hydrogen generation, and consequently yield a more conservative result than the TID-14844 release model.

Question No. H-48

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

Answer: Attached is a portion of the computer code output which is the analysis requested for the first ten days following the design basis LOCA.

INDIAN POINT NO. 2 ≠ TID RELEASE MODEL ≠ NOBLE GAS IN CORE

≠ PCT. ZIRC-WATER ≠ ALUMINUM INVENTORY PER SA ≠ C ≠ 148. SEPT. 18, 1969 11/20/69

TOTAL HYDROGEN GENERATION

DAY	TOTAL SUMP		TOTAL CORE		TOTAL CORROSION		GRAND TOTAL		
	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H2 SCF	RATE SCFM	TOTAL H_2 SCF	RATE SCFM	TOTAL H_2 SCF	VOL. PCT.
1	1.96E+00	4.15E+03	9.91E-01	2.06E+03	1.69E+00	3.41E+03	4.64E+00	1.63E+04	.73
2	1.43E+00	6.55E+03	8.39E-01	3.36E+03	8.72E-01	5.81E+03	3.14E+00	2.24E+04	1.00
3	1.18E+00	8.88E+03	7.60E-01	4.51E+03	8.72E-01	7.07E+03	2.76E+00	2.64E+04	1.18
4	9.26E-01	9.85E+03	7.10E-01	5.57E+03	8.72E-01	8.33E+03	2.51E+00	3.04E+04	1.35
5	7.82E-01	1.11E+04	6.74E-01	6.56E+03	5.36E-02	8.60E+03	1.51E+00	3.29E+04	1.46
6	6.72E-01	1.21E+04	6.45E-01	7.51E+03	5.36E-02	8.67E+03	1.37E+00	3.49E+04	1.55
7	5.87E-01	1.30E+04	6.20E-01	8.42E+03	5.36E-02	8.75E+03	1.26E+00	3.68E+04	1.63
8	5.17E-01	1.33E+04	5.98E-01	9.30E+03	5.36E-02	8.83E+03	1.17E+00	3.86E+04	1.70
9	4.62E-01	1.45E+04	5.78E-01	1.01E+04	5.36E-02	8.91E+03	1.09E+00	4.02E+04	1.77
10	4.16E-01	1.61E+04	5.60E-01	1.10E+04	5.36E-02	8.98E+03	1.03E+00	4.17E+04	1.84

Question No. H-49

Question: 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.

Answer: A copy of WCAP-7379L, Volume I, WCAP-7379, Volume II, and WCAP-7422L will be furnished separately. The copies of WCAP-7379L, Volume I and WCAP-7422L are non-proprietary versions of these referenced reports, the original versions of which are proprietary.

Question No. H-50

Question: 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.

Answer: The handling of spent fuel is performed entirely under water by remote means. It is physically impossible to lift the fuel assemblies out of the water because of the design of the fuel-handling crane and fuel-handling tools.

The Westinghouse analysis is significant since the solubility of iodine in the spent fuel pit water is not affected by the presence of other radioactive elements present as a result of a fuel-handling accident.

Question No. H-51

Question: 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.

Answer: All materials of construction in containment that would be exposed to the Design Basis Accident (DBA) environment were tested in solutions representative of the major constituents of the DBA environment chemistry. Only those materials that were compatible with the DBA environment were accepted for safeguards use, where these materials are exposed to the containment environment.

Question No. H-52

Question: 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.

Answer: The use of mercury and its compounds is prohibited in the reactor containment building. In addition, the use of mercury is prohibited on or in the vicinity of any system which has fluid contact paths with the reactor coolant or steam systems. All water used in the above systems is processed by flash evaporators and/or ion exchange resins for the purpose of removing contaminants. Chemical analyses are performed on the water on a routine basis for the purpose of varifying that the water meets the specified quality standards.

Question No. H-53

Question: Justify the use of test temperatures for aluminum corrosion 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.

Answer: The consequences of the postulated post-accident temperatures in the containment following a Design Basis Accident (DBA) were considered in the evaluation of aluminum corrosion in the alkaline borate spray solution. The data presented in Table Q6.3-5 of the FSAR are based on the aluminum corrosion data presented in Table Q6.3-4 and the DBA temperature profile curve (Figure Q6.3-2).

The use of aluminum in containment on engineered safety features is prohibited where the aluminum would be exposed to the containment environment.

Question No. H-54

Question: 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).

Answer: The Nordel rubber diaphragm valve material test results referred to in the question were the results obtained after the first examination of the test specimens which were exposed to a 33 w/o sodium hydroxide solution at a temperature of 100°F for a period of six (6) months. The test was not terminated at that point, but was extended for an additional 13½ months.

An examination of the test specimens after a total exposure time of 19½ months to the 33 w/o sodium hydroxide revealed that the specimen weight losses were <0.1% and that there was no change in the original physical characteristics of the Nordel rubber. These test results substantiated the original recommendation which specified the use of Nordel rubber diaphragm valves in the spray additive system.

Question No. 55

Question: 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).

Answer: AEC staff response.

Question No. 56

Question: 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 of 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 comtemplated 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 10CFR100 guideline values) as to the effectiveness of the ECCS?

Answer: a. Yes.

b. Yes

c. No

d. (AEC staff response).

e. The assumption of TID-14844 fission product release fractions implies degraded safety injection system operation which the applicant does not consider credible.

Question No. H-57

Question: 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 irrelevant? 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.

Answer: The test reports referred to in the FSAR were based upon all relevant test data and results known to the persons responsible for the preparation of such reports, and in reaching the conclusions in such reports, no known relevant information was ignored. It is possible that in determining relevant data for use in the reports, certain test results were disregarded as not appropriate. For example, test results, whether favorable or unfavorable, which may have been affected by faulty test equipment or procedure would have been disregarded. Thus, if a meter designed to read results were checked after a test had been run and shown to have been out of calibration, it is likely that the result from such test would have been disregarded. Similarly, in conducting detailed technical studies of the type involved in the test reports, the scientist or engineer may have disregarded certain tests, whether favorable or unfavorable, because the results of those tests would not bear on the phenomenon which was under study and which was to be the subject of the report.