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BEFORE THE UNITED STATES
ATOMIC ENERGY COMMISSION

trans w/ltr 3-12-71

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

OTHR

Answers of Applicant to Questions Raised

by Atomic Safety and Licensing Board

on January 19, 1971

Part I

March 11, 1971

KEY TO IDENTIFICATION OF QUESTIONS

(B) Question by Mr. Briggs

(G) Question by Dr. Geyer

(J) Question by Mr. Jensch

(Tr. 483) - Transcript Page 483

Question No. 1 (B) (Tr. 483)

"I find in the staff summary statements to the effect that the results of the Environmental Monitoring Program which has been conducted at the Indian Point for several years has shown no effect or that the releases of radioactivity have had no effect on the enfironment.

"I find similar statements in the applicant's summary and other reports, yet I find no evidence to this effect. It seems to me that since there now has been a considerable amount of experience in this area with measuring background, measuring the radiation levels and the other effects from the plant in operation, that it would be worthwhile and important to summarize this information in such a way that it is quite obvious to the person who reviews the summary that there have, in fact, been no detectable effects or what these detectable effects have been."

Answer:

There have been three relatively extensive sets of environmental analysis made in our case. Con Edison has maintained an Environmental Monitoring Program since 1958 in the vicinity of Indian Point; the State of New York has maintained Environmental Surveillance in the vicinity of Indian Point for almost ten years; and the New York University Institute of Environmental Medicine has conducted quantitative studies of radionuclides in Hudson River water, sediments, and biota since 1963.

The Environmental Monitoring Program conducted by Con Edison generally monitors gross amounts of beta activity in a variety of environmental samples. Were any large increase observed in the normal levels present, it would then be necessary to make specific analysis for the radioactivity present, to assess the possible dosemetric implications.

Some additional evaluation of radionuclides is made whenever gross activity measurements suggest the presence of unusual and unexpected amounts of activity.

To properly assess effects of radioactivity on man or the environment it is necessary to know the dose delivered either to man or to biota as a result of releases of radioactivity. Often the measurements of the activities in environmental samples is confused with an effect. Accordingly the only proper way to assess the effects of radiation release into the environment is to establish a radiation dose to man or to biota associated with the release. In assessing radiation dose, it is generally necessary to measure the radionuclide content of properly selected samples, and then to infer radiation dose from a knowledge of their radionuclide content .

The attached table lists radiation dosages to individuals residing near Indian Point for 1969. The year 1969 has been chosen because it was the year for which the highest liquid and gaseous releases have occurred to date from the plant. The dose to a nearby resident is so small that it cannot be measured directly. The dose was inferred from the measurement of gaseous activity at the release point and a knowledge of the meteorological dispersion. Dose from consumption of fish was calculated based upon measurements of fish by New York University. For purposes of making this estimate, a fish intake 50% higher than the United States average was assumed. The dose from plant operation to an individual living near the plant boundary with a substantial intake of Hudson River fish was about 0.4 millirems per year, about 1/2000 of the variability of natural background in the area, and 1/10,000 that at the permissible limits.

The highest dose to biota in the river from releases at Indian Point Unit No. 1 wasabout 120 mrem/yr to benthic organisms completely submerged in sediments. Fish received a smaller dose, less than 2 millirads per year. Acquatic vegetation which concentrates activation products well above levels found in fish, received a maximum dose of about 0.7 mrem/yr from Mn-54.

Radiation Dosages to Individuals

Residing Near Indian Point (3)

		mrem/year		
		minimum	mean	maximum
Measured Natural Background:		30	30	30
EXTERNAL:	Cosmic			
	Terrestrial	40	64	125
INTERNAL:		30	30	30
	Total	100	124	185

Calculated Increment from Indian Point Unit No. 1 Reactor (1969):

Gaseous Releases (1)	0.013
Liquid Releases (2)	0.030
Total	0.043

(1) Calculation based on 1969 gaseous releases.

(2) Calculation based on eating 30 grams fish/day using conservative model.

(3) See Question 11.1 of Indian Point Unit No. 2 FSAR for greater detail.

2. (B) (Tr 484)

:ON: "However, in looking at the technical specifications, I see many places where it says documents for inspection are not presently available and if such methods are developed that these inspections would take place, I would like to have information concerning what changes were made in the design of the plant or what provisions were incorporated in the detailed design of the plant for making the in-service inspection, what work was done by the applicant between 1966 and the present time to make these inspections possible, what programs the applicant will continue beyond the present date to make these inspections possible and what the schedules are for the completion of these programs." *

ER: The following areas within the reactor coolant system pressure boundary are available for visual examination and non-destructive testing:

1) Reactor Vessel - The entire inside surface
2) Reactor Vessel Nozzles - The entire inside surface.
3) Closure Head - The entire inside and outside surface.
4) Reactor Vessel Studs, Nuts and Washers.
5) Field Welds between the Reactor Vessel, Steam Generators, and Reactor Coolant Pumps and the Main Coolant Piping.
6) Reactor Internals
7) Reactor Vessel Flange Seal Surface
8) Fuel Assemblies
9) Rod Cluster Control Assemblies
10) Control Rod Drive Shafts
11) Control Rod Drive Mechanism Assemblies
12) Main Coolant Pipe External Surfaces (except for the foot penetration of the primary shield)
13) Steam Generator - The external surface, the internal surfaces of the steam drum, and channel head.
14) Pressurizer - The internal and external surfaces.
15) Reactor Coolant Pump - The external surfaces, motor and Impeller.

* "As I look at the technical specifications there are several places that indicate that inspections will take place 10 years from now."

The following design considerations have been incorporated in order to facilitate the above inspections:

1) All reactor internals are completely removable. The Tools and storage space required to permit these inspections are provided.

2) The closure head is stored dry on the reactor operating deck during refueling to facilitate direct visual inspection.

3) All reactor vessel studs, nuts and washers are removed to dry storage during refueling.

4) Removable plugs are provided in the primary shield just above the coolant nozzles, and in the insulation covering the nozzle welds is readily removable.

5) Access holes are provided in the lower internals barrel flange to allow remote access to the reactor vessel internal surfaces between the flange and the nozzles without removal of the internals.

6) A removable plug is provided in the lower core support plate to allow access for inspection of the bottom head without removal of the lower internals.

7) The storage stand provided for storage of the internals allows for inspection access to both the inside and outside of the internals.

8) The station provided for changeout of control rod clusters from one fuel assembly to another is specially designed to allow inspection of both fuel assemblies and control rod clusters.

9) The control rod mechanism is specially designed to allow removal of the mechanism assembly from the reactor vessel head.

10) Manways are provided in the steam generator, steam drum and channel head to allow access for internal inspection.

11) A manway is provided in the pressurizer top head to allow access for internal inspection.

12) All insulation on primary system component areas required to be inspected is removable.

Q. 2 (B) (Tr.484) Cont'd.

The proposed technical specifications indicated two areas where uncertain test results were anticipated because of material or geometrical considerations. Two of these areas include the steam generator tube sheet to head weld and steam generator safe ends. These areas proved to be inspectable during pre-service examinations.

The proposed Technical Specifications identify three areas in the reactor vessel for which remote inspection equipment must be developed. These areas are described in Items 1.1, 1.2 and 1.3 of Section 4.2.5 of the proposed Technical Specifications. A remote inspection system will be fitted to the Indian Point 2 plant within the ten years allowed by the code. Combustion Engineering, Babcock and Wilcox, Westinghouse and Southwest Research Institute are currently engaged in programs to establish procedures and techniques for remote inspection.

Southwest Research has already performed remote-automatic ultrasonic examinations of two reactors, one foreign and one domestic. The apparatus was custom built and procedures and methods were individually developed.

The proposed technical specification statement that some inspections would take place 10 years after initial operation stems from the inspection interval established by ASME Section XI. The code allows many components to be examined at or near the end of the inspection interval.

3. (B) (Tr. 485)

Question No. 3. "Also, I believe there is an indication that some, I will call it background information, must be available. Some information on the condition of the welds at the present time for use in comparison with measurements that are to be made in the future.

"I would like to have an indication of what this background information will be and how it is to be obtained prior to operation of the plant, if it is necessary that it be obtained prior to operation of the plant."

Answer:

Background information or base line data will be available for areas to be examined subsequent to plant operation.

ASME Section XI specifies that a pre-operational examination should be performed and the data therefrom should become the reference for all future post-operational examinations. Most of this pre-service inspection has been performed for Indian Point Unit 2. The examination methods are as specified in Section 4.2 of the proposed technical specifications.

Detailed procedures have been developed which specify the locations and methods of examinations. The procedures identify the particular test techniques to be utilized and data sheets to record the ultrasonic indications for the particular item being tested. A record of these indications can be used for future comparison purposes. These procedures have been devised to allow subsequent examinations to repeat the pre-service conditions.

Included in the pre-service examinations is a map of the Ultrasonic test results of the reactor vessel, performed after the hydro test which included the following areas:

3. (B) (Tr. 485) Cont'd

a) Vessel flange radius, including the vessel flange to upper shell weld.

b) Middle shell course

c) Lower shell course above the radial core supports

d) Exterior surface of the closure head from the flange knuckle to the cooling shroud.

e) Nozzle to upper shell weld

f) Middle shell to lower shell weld

g) Upper shell to middle shell weld

Question No. 4 (B) (Tr. 486)

"As I recall the staff answered this question rather briefly that the statement was made that Wash 740 was irrelevant to the present consideration and there was some small discussion of this.

I would like to ask that the staff look again at Report Wash 740, at TID-14844, and to tell again whether these two reports are irrelevant, if they are, why; if they are not, what has changed since the time of these reports to make the situation different from what was reported.

Answer

AEC Staff Response

5. (G) (Tr. 487)

Question #5:

My first question has to do with environmental monitoring, and in the Consolidated Edison Company's report on the environmental impact of Indian Point Station Nuclear Unit No. 2 there is a figure 17 which shows the location of numerous thermal dosimeters. I want to ask about these; what they record, how often they are read, what their full purpose is:"

Answer:

These dosimeters detect and integrate background gamma and cosmic radiation along with any gamma radiation from the plant. They are read monthly to determine if any change is occurring in the background radiation.

Question No. 6 (G) (Tr. 487)

"Also, I would like to find out more about the continuous monitoring system, just where the sensors are located, how much redundancy there is, what kind of alarms they sound and in connection with the discovery of unusual radiation, what provisions are made for warning the public, who makes the decision as to whether the public should be warned.)

Answer:

There are several types of samples which are taken continuously at various points outside the plant as part of the environmental monitoring program. While these samples are collected continuously they are analyzed on a weekly or monthly basis. These sampling systems have no redundancy, except insofar as there are several sampling points for some types of measurements. There are no alarms associated with any of these samples. These sampling points are described in attached figures 6-1, 6-2 and 6-3.

In addition to the above described continuous sampling systems there are two monitors external to the plant which provide continuous measurement. These monitors are:

1. An air particulate monitor at a point 800 feet southwest of the Unit No. 1 stack. If the radioactivity in the air exceeds normal levels, an alarm is indicated to the central control room operator.
2. A discharge canal monitor which likewise indicates an alarm to the control room operator if levels of radioactivity in the canal water approach limits.

Neither of these monitors have any redundancy.

As stated in the answer to Board questions No. 14, reliance will be placed upon in-plant instrumentation, not the above described out-of-plant instrumentation, in making the initial decision as to whether the public should be warned in the event of unusual radiation from a radiation accident. The New York State Health Department would make the decision whether the public should be warned. In accordance with the State's emergency plan, if the Health Department determined such warning to be necessary, the Department would promptly disseminate information to the affected public on recommended protective action by the most expeditious means available. We understand that the State Health Department would use the facilities of the State Civil Defense Commission, police and fire departments, radio and television and other available means as appropriate.

CONTINUOUS [illegible] MONITORING STATIONS

AT INDIAN POINT

Station No.	Media	Type	Sampling Frequency	Method of Collection	Locations	Analysis	Minimum Sensitivities	Measurement Instrumentation	Remarks
1	Fallout	Continuous	Monthly	Open pot type collector	Point 1 and 15 miles south of site of Eastview	Gross beta and tritium	1 picocurie per liter for gross beta 3000 picocuries per liter for tritium	Gas flow, windowless proportional counter for gross beta Nuclear Measurement Corporation Type PC 3A Type PC 11A Type PC 11T	Measurements made 48 hours after collection to allow for decay of randonthoron daughters
2	Air Particulate and Organic Iodide	Continuous at 1 CFM	Weekly	Two fixed membrane filters (0.8 micron size) preceding a charcoal filter	Points 1, 2, 3, 4 & 5 and in addition offsite at points in Peekskill, Buchanan, and Verplanck for one week periods consecutively	Gross beta and gamma spectrum	0.1 picocuries per cubic meter for gross beta	Same as 1 for gross beta	Measurements made soon after collection and 48 hours later to allow fo[r] decay of randonthoron daughters
4	Hudson River Water	Continuous	Weekly	Continuous flow regulated to fill 50 gal. drums. Representative sample taken once a week and drums emptied	Hudson River inlet pipe into the plant, and at plant discharge canal. Points 9 and 10	Same as 1 and tritium on monthly composite	Same as 1	Same as 1	Same as 1
15	Direct Gamma	Continuous	Monthly		Selected locations in Buchanan, Verplanck Montrose, Peekskill, and at a number of points on-site at the plant perimeter	Gross Gamma background	1 mr	Victoreen Ionization Chamber Model 239 0-10 mr or Film Badges	
	Direct Gamma	Continuous	Monthly	-----	Eleven site locations shown on Fig. 17	Same as 15	Reportedly sensitive to very small changes in gamma radiation	Thermoluminescent Dosimeters	Installed on trial basi[s]. Sensitivity and reproducibility under evaluation

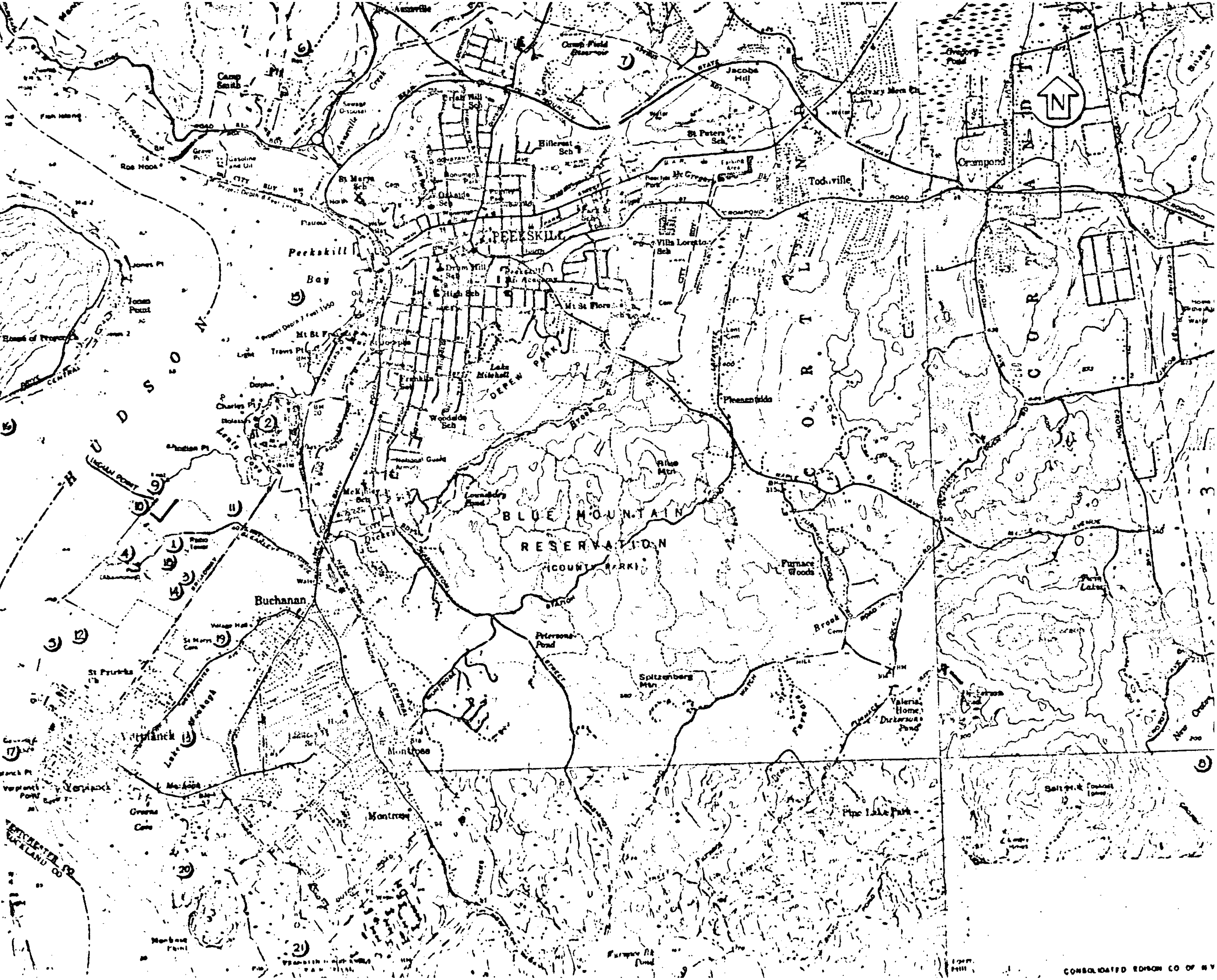
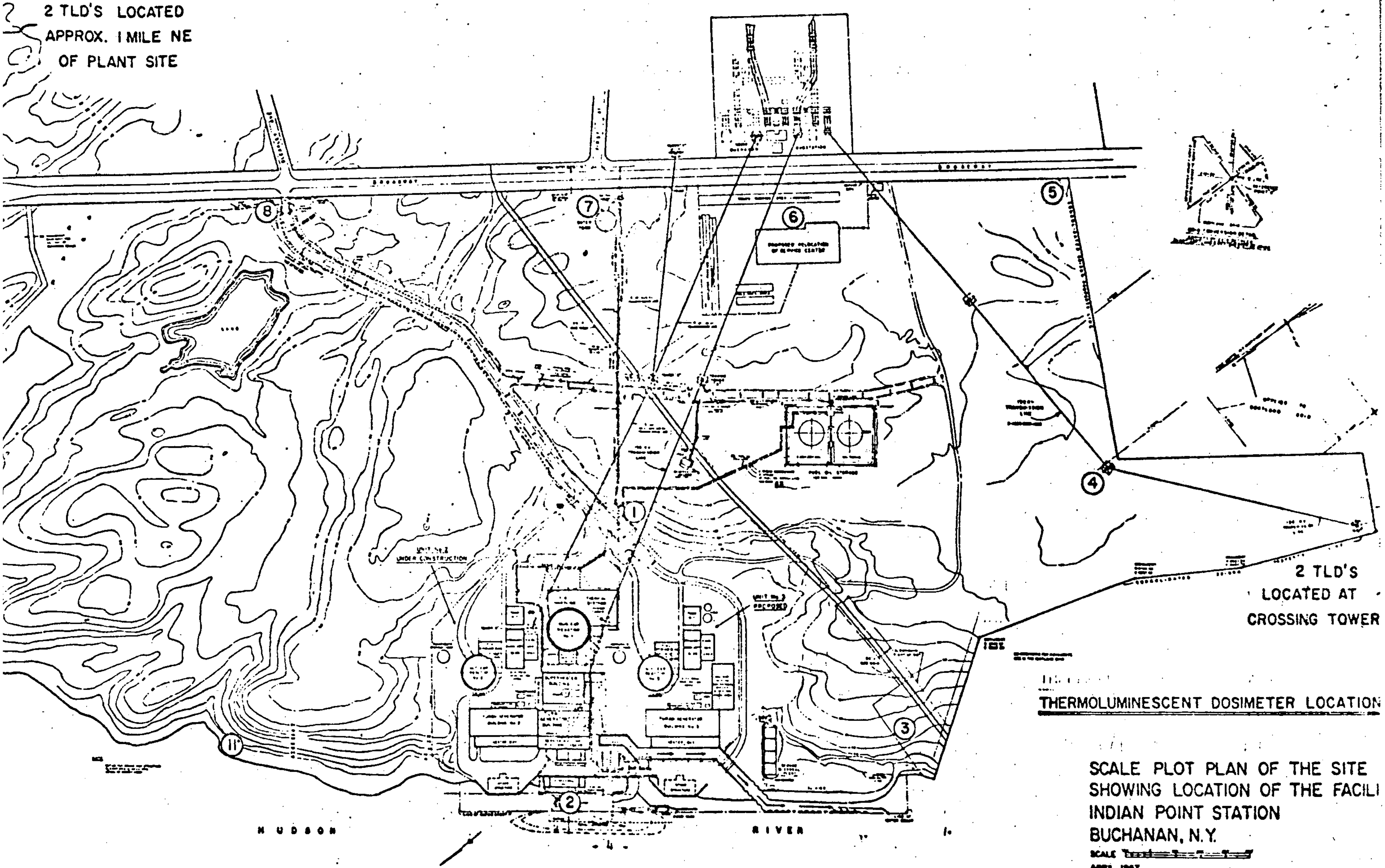


Figure 6-2

(G) (Tr. 487)



THERMOLUMINESCENT DOSIMETER LOCATION

SCALE PLOT PLAN OF THE SITE
SHOWING LOCATION OF THE FACILI
INDIAN POINT STATION
BUCHANAN, N.Y.
APRIL 1967

<u>Question No. 7</u> (G) (Tr. 487)

"In connection with the monitoring program it would be interesting to know if any consideration has been given to daily publication of radiation levels in the region just as they now report weather or air pollution levels or pollen counts. They might assure the public to see what goes on continuously.

<u>Answer</u>:

Applicant believes that the decision whether to make daily publication of radiation levels should be made by a responsible government agency. However, itis applicant's view that publication of radiation levels would not be useful to the general public. The variation in measured natural background radiation levels from one location to another in the vicinity of Indian Point is considerably greater than the smaller increment from the Indian Point plant. Daily publication of variation of these background levels would not provide the general public with meaningful or useful information such as is the case with information on weather, air pollution and pollen counts.

Question No. 8 (G) (Tr. 487)

"In connection with Dr. Brigg's question about WASH 740, the whole problem, a very complex problem of risk versus benefit versus cost in connection with these environmental matters has been brought up in discussions earlier in this hearing. It might be interesting to hear the staff in particular addressing itself to how it considers this problem."

Answer:

AEC staff response.

Question No. 9 (G) (Tr. 488)

"Other areas of interest are the question of the burnable poison that has now been designed into this reactor, how it is fastened in, how it functioned, what experience there has been with such burnable poison, what assurance is there that it is going to be there when needed."

Answer

Each fuel assembly contains 21 steel thimbles which replace fuel rods in the lattice. 20 of these thimbles guide the control rod pins through the assembly when the assembly is in a control rod position. The remaining thimble is used for the moveable flux detectors which also may pass through the assembly. In the first cycle of core operation when the core is more reactive than in later cycles, burnable poison pins are placed in most of the assemblies which are not at control rod positions. Their purpose is to reduce the concentration of chemical shim in the critical core at the beginning of the first cycle. Without the burnable poison, a higher soluble poison concentration would be required and a positive moderator temperature coefficient would result due to the expansion of water carrying dissolved chemical poison out of the core.

All the burnable poison pins for one fuel assembly are screwed and welded to a holddown plate which is held in position under the upper core plate. The burnable poison pins slide into the control rod guide thimbles and the complete poison assembly, consisting of 8, 12 or 16 pins fixed to a holddown plate, is loaded into appropriate assemblies at the fuel element factory.

Each pin consists of a steel tube containing a glass tube with an inner steel sleeve inside the glass tube. The glass contains 12.5% by weight of B_2O_3, the B-10 in the glass acts as a neutron absorber to reduce the initial reactivity of the core. As neutrons are absorbed, the B-10 depletes, roughly 10% is left at the end of the first cycle when all burnable poison rods are removed. The tube is completely sealed and the glass is supported by the inner sleeve and outer clad.

Identical burnable poison pins have been irradiated in the Saxton experimental reactor and are in use in the Beznau reactor, Switzerland, the R.E. Ginna reactor, the H. B. Robinson reactor and the Point Beach reactor where they have successfully performed their function which is to assure that the moderator coefficient is less than zero at operating conditions early in core life before the coefficient is made negative by core burnup. In two or three months, they will be removed from Beznau and R. E. Ginna since they are no longer needed to maintain a negative moderator coefficient.

Question No. 10 (G) (Tr. 488)

"Another question having to do with the internal safety features is the matter of crucibles beneath the reactor which is now a longer time than is desirable. It would be interesting to hear why this was considered desirable and what made it then considered to be unnecesaary."

Answer: As stated in the FSAR, there have been several design modifications incorporated into this plant for emergency core cooling since submission of the preliminary report and issuance of the construction permit. They are as follows:

1. Increased capacity of emergency core cooling.
2. Deletion of the reactor pit crucible.
3. Valving and piping modifications in the emergency core cooling system to give added assurance of core and containment cooling in the very unlikely event of a passive component failure during long-term cooling following a loss-of-coolant accident.

The increased capacity of the emergency core cooling system results from the addition of a pressurized accumulator to each coolant loop which provide rapid core reflooding capability with borated water after a major loss of coolant accident. As a result of the increased cooling system capacity, clad melting is effectively prevented for rupture sizes up to and including the double-ended severance of a main reactor coolant pipe. The detailed analysis of such breaks is shown in Section 14.3.3.

In the prior design of the emergency core cooling system, core reflooding following a loss of coolant accident was accomplished by three high head and two high flow safety injection pumps and by the two high flow residual heat removal pumps. The reflooding rates with this design were not sufficient to prevent the fuel clad temperature on the highest power fuel rods from rising to the clad melting temperature, hypothetically assuming instantaneous severance of a coolant loop. Further, the additional pumping capacities and emergency power requirements necessary to provide reflooding times that would not result in clad melting for a loop severance were prohibitively large. Because of this and the uncertainties involved in demonstrating that the fuel pellets released from the melted clad could not fall to the bottom of the reactor vessel a provision was proposed for containing the melted fuel in this unlikely event. This provision was a refractory lined crucible to be located directly beneath the reactor vessel. Extensive research and development efforts were initiated in the areas of: 1) designing a core reflood system that would limit clad temperature to below melting and 2) the design of a crucible that could contain the molten core. Because of the success in developing a highly improved core reflooding system and the continuing uncertainties associated with the behavior and containment of molten fuel, more stringent core cooling criteria were adopted to preclude fuel clad melting and prevent significant clad water reaction and hence insure the preservation of the core heat transfer geometry. The increased capability of the emergency core cooling system to meet these new criteria are reflected in the design by the inclusion of four pressurized accumulator tanks containing a large volume of borated water held back from the reactor system by check valves which open (without requiring a signal) to discharge into the reactor coolant system when the system pressure decrease associated with a loss of coolant falls below their discharge pressure. These four accumulator tanks supplement the two high flow safety injection pumps. The rapid water discharge from these accumulators greatly reduces the core reflooding time thereby supplying earlier core cooling and limiting the clad temperature increase to a value well below the melting temperature. As a measure of effectiveness of the accumulators, the core midplane reflood time after a loop severance is less than 35 seconds with the revised design as compared to about 300 seconds with the initial design assuming one high head and one low head pump ineffective in both cases. This direct approach of reducing the core reflood times and retaining the core intact eliminates the problem of containing the fuel pellets and the possibility of core migration and thus the need for the reactor pit crucible associated with the slower reflooding rates provided by the initial emergency core cooling system design. Hence, the reactor pit crucible has been deleted from the plant design. Details of the design of the revised emergency core cooling system including the accumulators are presented in Section 6.2. A complete analysis of the capability of the revised emergency core cooling system to accomodate the loss of coolant accidents, including supporting basis, assumptions and results which show that the new emergency core cooling system design meets the revised criteria is included in Section 14.3.

The valving and piping modifications in the emergency core cooling system give capability to maintain core cooling and containment cooling in the event of a passive component failure in the safety injection system or service water system for the long term after a loss of coolant. The design also has sufficient component redundancy to accomodate an active component failure.

Question No. 11 (G) (Tr. 488) ASLB 1/19

"Finally, in the earlier discussions there were references to an accident at Indian Point that produced high fallout at Yorktown. Now, we have no evidence on this so far as to just what did happen, but it would be nice to clear this matter up, and if there was such an occurrence, what did it amount to and why was this statement made?"

Answer:

There was no accident, or accidental or abnormally high release of radioactivity at Indian Point on or about May 18, 1970. There is no connection between the May 18, 1970 Croton reservoir reading and operations at Indian Point.

As for the Yorktown reading (Croton Reservoir), the fact that, as Mrs. Weik says, it was measured nowhere else in the State of New York implies that there may have been some error in measurement. For further elaboration on the unusual reading at the Croton Reservoir, the following direct quotation from the State of New York's Department of Environmental Radiation Bulleton 70-2, October 5, 1970, is provided:

"A grab water sample collected on May 18, 1970 from Croton Reservoir at Taconic showed a gross beta of 80 pCi/l. An isotopic gamma analysis was made on this sample and ruthenium-106 was non-detectable, and zirconium-95 was 53 pCi/l. An Algae sample was collected July 9, 1970 at the same sampling point and results were as follows:

RuRh-106	2,816 pCi/kg	ZrNb-95	1,484 pCi/kg
Cs-137	479 pCi/kg	Co-60	non-detectable

Gross beta results of grab samples taken from this same sampling point around the period of the relatively high result are given below:

4/16/70 -	5 pCi/l	7/ 2/70 -	4 pCi/l
5/18/70 -	80 p Ci/l	7/15/70 -	4 pCi/l
6/16/70 -	7 pCi/l		

It was concluded that the water sample with the high result was collected too close to the shoreline in shallow water and some algae was included in the water sample. The radioactivity found in the water sample and in the algae sample appears to have originated from fallout associated with atmospheric weapons testing. This sampling point has been changed to deeper water in the Croton Reservoir in order to obtain a more representative water sample in the future."

The following is a quotation from the Indian Point Station Semi-Annual Operation Report No. 16 covering the period April 1, 1970 to September 30, 1970, which explains the cause of the plant shutdown referred to by Mrs. Weik:

> "Following a two month refueling outage, Unit No. 1 was returned to service on May 20, 1970 with primary loops Nos. 11, 12 and 13 operating. Loop No. 14 was isolated due to a tube leak in its associated boiler which developed on May 16, 1970 during a hydrostatic test of the primary system. Within a few hours after the Unit had been placed in service, a primary to secondary leak was detected in No. 12 nuclear boiler. The unit was shut down at 9:55 P.M. on May 20, 1970 in order to locate and plug the tube leaks in Nos. 12 and 14 boilers."

Question No. 12 (B) (Tr. 488)

"In reviewing the reports, a question on the detail came to mind. The question came to mind as a result of an experience back in the middle 140s that occurred many times before June of 1946, and I assume it has happened since. It has to do with the use of transit as a fire barrier.

Before the mid-40s, it was used as a fire barrier and the temperature when it got up as much as 500 degrees Fahrenheit, the transit could be expected to explode.

I see in the report it is used in aeration of control wiring and power wiring. I would like to have some information concerning changes that have been made in the transit since the middle '40s that make this procedure useful. Also whether this characteristic of transit was concerned in specifying the material for the fire barriers."

Answer

"Transite" is an exclusive trade name for a non-laminated asbestos-cement product manufactured by Johns-Manville. Johns-Manville has indicated that they are not aware of any tests and resultant explosions of "Transite" during the 1940s. They suggest that the actual material tested at that time was a laminated asbestos-cement product made by another manufacturer and mistakenly referred to as "Transite."

The Johns-Manville "Transite" used in Indian Point Unit No. 2 is made by a press process which results in a homogeneous structure. Most other asbestos-cement products are manufactured in a way which results in a non-homogeneous laminated structure with high moisture content. Exposure of such non-homogeneous products to temperatures of 212°F and higher can form steam and the rapid increase in pressure causes explosive delamination.

Corrugated "Transite", manufactured in a similar fashion to flat "Transite" did not explode on testing. This was true for fire or oven exposure of "Transite" completely saturated with water. Similarly, immersion of "Transite" while at elevated temperature did not result in explosion. Identical tests were performed on non-homogeneous material manufactured by other processes and that material exploded.

"Dr. Geyer mentioned the elimination of the crucible. There is a statement made in the report that although the crucible has been eliminated, that provision has been made in the insulation so that water has access to the bottom of the reactor vessel and I assume that means the water would provide some cooling for the bottom of the reactor vessel.

I would like to have information concerning how effective this can be expected to be, what sort of conditions it would take care of, and what certainty there is that water will have access and will in fact cover the bottom of the reactor vessel under accident conditions."

Answer: Preliminary calculations made during the conceptual design of the crucible indicated that water level around the reactor vessel would cover the bottom of the vessel and if in contact with the vessel, the water could provide adequate cooling so that it might be expected that molten fuel could be contained by the bottom head of the vessel. Accordingly, the reactor vessel insulation was designed to permit the water to contact the vessel surface.

The efficacy of this cooling mode, like the other phenomena related to behavior of molten fuel, could not be well defined because of inability to simulate the system experimentally. Hence the decision was made to upgrade the emergency core cooling system and obviate the need to design molten fuel entrapment and cooling.

13. (B) (Tr. 489) ASLB 1/19

The design feature of the vessel insulation referred to above was retained, as it did not interfere with the other functional requirements of the insulation.

<u>Question No. 14</u> (B) (Tr. 489)

"...the reading I have done so far gives me the impression that if there were an accident and an accompanying considerable release of radioactivity, that the applicant is responsible only for notifying the State of New York and other agencies that this has occurred and the provisions that must be made for taking care of the public after that are the responsibility of those agencies.

"I would like to have some information concerning the negotiations that have been taking place or have taken place between the applicant and the various public agencies concerning the emergency procedures, the procedures that can be expected to be used and where the responsibility lies in the event of serious accident."

<u>Answer</u>:

Emergency Plans for Indian Point Unit No. 2 describing the activities of Con Edison and the notifications to be made by Con Edison, including requests for assistance, are described in the response to FSAR Question 12.5 in the section titled "Radiation Contingency Plan." Within this Plan are three different categories which may require varying degrees of implementation of protective actions, described beginning on page 6 under Section 4.2 titled "Implementation Levels." The first category is the local contingency plan which primarily would involve a potential for the need to take protective actions within the site boundary. Also described is the site contingency plan which involves a potential that may require protective actions beyond the site boundary. The third category is titled general contingency plan which involves a site contingency for which the off-site effects have been verified by monitoring and surveys off-site.

Con Edison's Radiation Contingency Plan requires that notification be given to the AEC's New York Operations Office and to the New York State and Westchester County Departments of Health that a site contingency has been declared. This would be prior to the declaration of the general contingency, which would not be made until off-site monitoring by Con Edison had taken place. These requirements for notification are described on page 26 of the Radiation Contingency Plan.

Although it is exceedingly improbable that an accident will occur at Indian Point Unit 2 which will require protective actions off-site, over the past several years Con Edison has held numerous meetings concerning the Radiation Contingency Plan with various representatives of the State of New York, the New York Operations Office of the AEC and the Westchester County Health Department. Actions of State agencies in response to a major nuclear accident are described in New York State's emergency plan for major radiation accidents.

The State's emergency plan describes the criteria for determining whether protective actions are needed, the protective actions to be considered to minimize public exposure to radiation and the authority and responsibilities of the various officials and agencies involved. It further provides for appropriate public announcements.

In accordance with the State's plan, the State Department of Health, upon notification from Con Edison that a site contingency had been declared, would determine the necessity for protective actions off-site and direct the various actions required.

Con Edison has discussed general procedures to be followed and the information that should be provided in the event of a site contingency with the Department of Health and various other State agencies involved in the State's emergency plan. The Department of Health indicated its desire to consider the need for protective actions at the earliest moment following the onset of a serious accident, rather thanwaiting for off-site monitoring results to confirm the magnitude of any accident which had taken place. To this end, the Department of Health has requested, and Con Edison has agreed, that in the event of a site contingency, Con Edison will notify the Department of Health through the officer on duty at the 24-hour emergency number of the New York State Civil Defense Commission warning point located near Albany. Con Edison will also provide the following information: the type of accident that has occurred; the safeguards which are effective; gross activity levels inside containment as determined by gross gamma instrumentation which observes containment activity through steam line beam holes; a statement as to the nature of the release to the containment; wind speed; wind direction and meteorological category.

Con Edison will further provide the Health Department with calculated thyroid dose levels due to iodine 131 at various distances downwind based upon the activity within containment and an assumed 1/10 of a percent per day leakage from containment. The 1/10 of a percent per day leak rate from containment is assumed even though the pressurized weld channels and penetration system along with the seal water injection system is designed to prevent such containment leak rate because the field survey monitoring which would verify that such containment leakage is not occurring would not yet be available on this initial notification. If means are available of verifying that containment leakage is not occurring at the time of the initial notification or that it is considerably below the 1/10 of a percent per day assumed, the calculated doses will be adjusted accordingly.

The State Health Department has indicated to Con Edison that these dose estimates will be used by the Department as a primary tool in making the initial determination as to which, if any, protective action should be implemented immediately. Subsequently, off-site monitoring data will provide the necessary information concerning off-site releases upon which the Department's determination of the need and desirability of subsequent protective actions would be based. The potential for significant off-site releases could exist only if the containment inventory of iodine were to be far in excess of the amount anticipated. In this connection, the redundant core cooling features are designed to limit the iodine inventory to that released from the gap.

As previously indicated, Con Edison will notify the AEC's New York Operations Office and the Westchester County Department of Health of a site contingency at the same time as the State Health Department. Within the AEC's New York Operations Office there is a radiological assistance team under the direction of an AEC group leader, which team consists of local AEC personnel equipped with appropriate survey instruments who will be able to assist in monitoring the effects of radiation releases from the site. We are advised that the Westchester County Department of Health would provide additional radiological survey supporting effort. Initial radiation surveys off-site would be by Con Edison's plant health physics survey team who would utilize a survey truck with 2-way radio communication to the central control room. These personnel would monitor airborne radioactivity and direct radiation downwind of the site in the event of a site contingency. There would also be available through the AEC interagency support from various other Federal agencies and national laboratories, under the Interagency Radiological Assistance Plan. These groups can provide additional trained personnel for monitoring and advisory activities to help support Con Edison's contingency plans and the State Health Department's activities.

Question No. 15 (B) (Tr. 490)

"The technical specifications indicate that the releases from the plant will be limited to those which will make certain that the public is not exposed to radiation levels above those provided in the 10CFR, Part 20 guidelines. We understand that the plant will normally operate with releases that are far below those guidelines.

Is there reason why the technical specifications contains no time limits on the releases to the 10CFR, Part 20 limit and should not such time limits be included in the technical specifications? I assume that the technical specifications were written by the applicant and that he has a certain amount of freedom in what he puts in the technical specifications, at least until the time they are accepted by the AEC."

Answer

Applicant's proposed Technical Specifications, Section 3.9, limits even the maximum instantaneous release rate to 10CFR20. The requirement in this specification to keep releases as low as practicable would require prompt correction of any condition causing higher than normal releases. (Normal releases are expected to be only a small fraction of 10CFR20). Therefore, a time limit on releases at 10CFR20 levels is not required.

Question No. 16 (J) (Tr. 490) ASLB 1/19

" Dr. Geyer referred in one part to the burnable poison and suggested that experimental test data might be of interest to confirm those conclusions with reference to burnable poison. I wonder also as a general matter if more of the experimental test data can be shown for several of the safety engineered components that are accepted in this proposal for this reactor.

"For instance, the emergency core cooling system, what are the data that confirm the conclusions in that regard? I know in previous cases this subject has come up, but it is referred to continuously as research matter and there may be data which is more updated than we have last considered and might give us a summary of the R&D in this regard."

Answer

The answer to this question will be forthcoming shortly.

Question No. 17 (J) (Tr. 491)

"Speaking of research and development, the Board is concerned concerning the reports issued by the Advisory Committee on Reactor Safeguards over a period of time in reference to pressurized water reactors, and I wonder if a summary can be presented of what those concerns are as having been expressed by the Advisory Committee on Reactor Safeguards over, say, the last ten years because the ACRS, and I refer to them as the Advisory Committee on Reactor Safeguards, concluded many of its reports by saying if these matters are carried out then there is reasonable assurance that the reactor can be operated without undue risk to health and safety of the public."

Answer

AEC staff response

Question No. 18 (J) Tr. 491

"Aside from a summary statement, or in addition, let me say, to a summary statement in that regard and updating of the experimental test data under those research and development projects, I wonder if we could have a witness from the staff of the Atomic Energy Commission about the research and development work. I think some boards in the past have had difficulty with summary statements maybe not being as complete as they would like to have it. If a witness is present then I think any further inquiry the Board may have can be readily considered and answered at that time.

For instance, as I recall it, there is a loss-of-fuel test. That has been going on for sometime, and maybe we can have some data about that and the other R&D programs that ACRS has outlines...

Are they carried on with the same vigor and financial support, for instance, that heretofore has been allocated to other projects and what has been discovered to date and what more is left to be done and when will that work be done and what is the data that is expected to be derived from further work in that regard?...

I think it is important that we have a witness from that work, a witness that has a responsible position.

Maybe it would be the director of the reactor development technology himself to participate in this hearing; I think it would be very helpful if he would."

Answer

AEC staff response

Question No. 19 (J) (Tr. 495)

"On page 113 of the detailed statement on environmental considerations by the staff... we find HEW's statement, something to this effect: The estimate of liquid radioactivity discharges and so forth, in our judgement, is not adequately documented.

What do they want in order to make the reviews? Did the staff get this to them? Is there anything further from HEW other than that which is reflected in the staff detailed environmental statement reflected on page 113?

In fact, is there any supplementary cost to any of the agencies to which the Applicant's statement is submitted?

Answer

AEC staff response

Question No. 20 (J) (Tr. 495)

"Then there is this further statement shown on page 113 of the staff detailed environmental statement which says something like this: Current PWR, I take that as "pressurized water reactors," operating experience indicated that both the liquid radioactive discharge and gaseous discharges will be considerably higher and the Applicant has not desired new design implications to support the lower effluent discharges. Can the staff give us what figures reflect the current PWR operating experience and indicate that both the liquid and gaseous discharges will be higher, higher than what, the Applicant considered, or what has been designed in other reactors and what kind of design information does HEW believe will be necessary for it to support or give a conclusion respecting the estimated lower discharges?

Answer

AEC staff response

Question No. 21 (J) (Tr. 496)

"On page 114 of that statement staff supplement there is the statement by a public health physician of HEW, the proposed technical specification for the site gaseous waste discharge limits would be excessive if calculated by the method indicated by the Applicant."

Answer

On November 12, 1970, Applicant responded to comments on Applicant's Environmental Report made by Federal agencies in a letter to Peter A. Morris, Director, Division of Reactor Licensing, Atomic Energy Commission from William J. Cahill, Jr., Vice President, Consolidated Edison Company of New York, Inc. As stated in that letter, "With respect to the site gaseous waste discharge limit, a typographical error appeared in the equation for the allowable gaseous release rate from the Indian Point site as first submitted to the AEC in the FSAR. Subsequent to the HEW review, the error was corrected and the equation rewritten to avoid misinterpretation. The correct equation is as follows:

$$\left(\frac{\chi}{Q}\right)_1 \sum_i \frac{Q_1 i}{(MPC)_i} + \left(\frac{\chi}{Q}\right)_2 \sum_i \frac{Q_2 i}{(MPC)_i} \leq 1.0$$

where:

i refers to any radioisotope.

Q_1i and Q_2i are the release rates (Ci/sec) of any radioisotope i from Unit No. 1 and Unit No. 2, respectively.

(MPC) is in units of µCi/cc as listed in Column 1, Table II of Appendix B, 10CFR20, except that for isotopes of iodine and particulates with half lives greater than eight days, the values of $(MPC)_i$ shall be reduced by a factor of 700.

The above specification applies to the entire Indian Point site and will be modified to accommodate Unit No. 3 when it is completed and in operation."

Question No. 22 (J) (Tr. 496)

"HEW also said discharge limits for Indian Point facility should also be applied for Con Ed Units 4 and 5 if these additional units were built at the proposed location about 1500 meters south of the Indian Point site."

Answer

Con Edison has already indicated that Indian Point Units 1, 2 and 3 should be treated as a single facility in establishing discharge limits. Nuclear Units 4 and 5 are not under review in this context, however, this comment by HEW will be taken into consideration in the licensing review of Nuclear Units 4 and 5 (Verplanck 1 and 2).

Question No. 23 (J) (Tr. 496) ASLB 1/19

"The statement is also made the environmental surveillance program for the facility would be adequate if modified to include the LDs, and I take it that is total limitation doses with the minimum sensitivity of a dash 10 millirems per month."

Answer

We are evaluating the use of thermoluminescent dosimeters now and expect that it will be possible to measure doses about 10 millirems per month with them.

Question No. 24 (J) (Tr. 496)

"The suggestion is made by HEW on page 115 of the staff's submittal, estimates for gaseous releases for Indian Point No. 2 were based upon a 45 day holdout. We believe the capacity should be expanded to 60 days and it comments further:"

Answer

With respect to radioactive waste treatment and holdup systems, the revised proposed technical specification and bases for Indian Point Unit No. 2 (Specification 3.9 Effluent Release) which was submitted to the AEC subsequent to the HEW review, contains the following commitment:

> "Plant equipment shall be used in conjunction with developed operating procedures to maintain surveillance of radioactive gaseous and liquid effluents produced during normal reactor operations and expected operational occurrences in an effort to maintain radioactive releases to unrestricted areas as low as practicable."

HEW suggested that the gaseous waste holdup capacity should be expanded to 60 days minimum. The final technical specification required a minimum of 20 days holdup in the gas decay tanks, except for low radioactivity gaseous waste resulting from operations associated with refueling and startup. The design capacity of the tanks allows a 40 day holdup based on design flow rates. Variation in those rates may permit a longer holdup time. However, the 20 day minimum required by the technical specifications result in discharges that constitute a small percentage of maximum permissable concentrations.

25. (J) (Tr. 497)

"Apparently the position taken by HEW is said to be taken because gaseous releases during normal operating at Indian Point Unit No. 1 have been much higher than at other similar operating PWR's which could be interpreted to indicate that the gaseous waste holdup was not used to the fullest extent, and so forth.

Could the staff get those figures or could the Applicant? What were the releases from Indian Point No. 1 which were higher than other similar operating PWRs? What are other similar PWR's and what were the figures for releases from them?"

Answer

AEC staff response

Question No. 26 (J) (Tr. 497)

"Incidentally, in considering what the releases are from Indian Point Unit No. 1 and other PWRs, especially in New York State, can those readings be compared with the readings of the environmental surveillance undertaken by New York State monitoring groups? What are their figures?...

We aren't so worried about the conclusions if the figures are shown and we would like to see the figures."

Answer

See NYS Department of Health and NYS Department of Environmental Conservation Environmental Radiation Surveys from 1959 to 1969, which will be submitted separately.

Question No. 27 (J) (Tr. 498)

"There was mention made, I believe, by Dr. Briggs about TID-14844. I wonder if we could have a computation precisely in accordance with TID-14844, together with the components, other components of that calculation.

I understand that they have used some TID-14844 and some other components which I think are justified; but I think we should start with 14844 and give us that from both the staff and the Applicant because as I understand, TID-14844 is a guideline that can be applied until other engineering data are shown to justify variance therefrom and there may well be engineering data in that regard but if we can start from the beginning point, that would help us to evaluate the safety considerations of the engineering matters that seem to justify a variance."

Answer

See Table 27-1, with Attachments #1 and #2.

THYROID DOSE - REM

TABLE 27-1

CASE #		SPRAYS* NONE	SPRAYS* AEC**	SPRAYS* CON ED***	FILTERS* NONE	FILTERS* AEC**	FILTERS* CON ED***	CONTAINMENT* LEAK RATE TID 14844	CONTAINMENT* LEAK RATE AEC**	SOURCE* TID 14844 A	SOURCE* TID 14844 B	SOURCE* GAP ACTIVITY A	SOURCE* GAP ACTIVITY B	METEOROLOGY* TID 14844	METEOROLOGY* CON ED	DOSE (REM) 2 HRS. @ 520m.	DOSE (REM) 30 DAYS @ 1100 m.
1		x			x			x		x				x		1,210	30,400
2		x			x				x	x				x		1,210	17,200
3		x			x			x			x			x		1,210	30,400
4		x			x			x				x		x		80	3,370
5		x			x			x					x	x		80	3,370
6		x			x				x		x			x		1,210	17,200
7		x				x		x		x				x		775	[illegible]
8		x				x		x			x			x		813	[illegible]
9		x					x	x		x				x		769	573
10		x					x	x			x			x		777	558
11			x		x			x		x				x		138	63
12			x		x			x			x			x		245	3,100
13				x	x			x		x				x		20	10
14				x	x			x			x			x		138	3,050
15			x			x			x	x				x		125	61
16			x			x			x		x			x		228	360
17				x			x		x	x				x		19	9
18				x			x		x		x			x		102	80
19				x			x		x			x		x		1	1
20				x			x		x				x	x		7	5
21		x			x			x		x					x	1,210	3,500
22			x			x			x		x				x	228	219
23				x			x		x		x				x	102	65
24				x			x		x				x		x	7	4
25	TID 14844	x			x			x		x				x		1,210	30,300

Breathing Rates: Cases 1-24: $3.47 \times 10^{-4}\ m^3/sec$ @ 0 - 8 hours
$1.75 \times 10^{-4}\ m^3/sec$ @ 8 -24 "
$2.32 \times 10^{-4}\ m^3/sec$ @ 24 "
Case 25: $3.47 \times 10^{-4}\ m^3/sec$ @ 520 meters
$2.32 \times 10^{-4}\ m^3/sec$ @ 1100 meters

* See Details on attached Sheet #1.
** AEC Division of Reactor Licensing Safety Evaluation - IP-2, Nov. 16, 1970
*** One spray pump and 3 fans (5000 CFM each) operating.

Attachment #1.

Summary of Thyroid Dose Calculation Parameters.

I. Iodine Removal Constants:

		AEC	Con Ed
(i) Sprays:	Inorganic	$\lambda_s = 4.5$	$\lambda_s = 32.0$
	Organic	$\lambda_s = 0$	$\lambda_s = 0$
(ii)	Inorganic	$\lambda_{cf} = 0.49$	$\lambda_{cf} = 0.4985$
	Organic	$\lambda_{cf} = 0.048$	$\lambda_{cf} = 0.3877$

II Containment Leak Rate:
(i) TID 14844: 0.1% per day
(ii) AEC : 0.1% per day for the first day
0.05% per day thereafter

III Source:
(i) TID 14844: 25% of the iodine is available for release
(ii) GAP Activity: 3% of the equilibrium core I-131 inventory.

Case A assumes no organic iodine and Case B assumes 90% inorganic and 10% organic.

IV Meteorology:

(i) TID 14844: inversion type weather conditions.
(ii) Con Ed : three periods are considered:

1) First two hours after the accident--Inversion parameters of TID-14844 are assumed.

Category:	C_y	C_z	N	$\bar{u}$	X_o
Inversion-I	0.4	0.07	0.5	1 m/sec	430 m

2) Next 22 hours - The same inversion condition is assumed to exist, but the average wind speed is 2 m/sec.

3) From 1 to 30 days:

Category	Fraction	$1/\bar{u}$	C_z	C_y	n
Lapse-L1	0.137	0.575	0.48	0.6	0.2
Lapse-L2	0.061	0.191	0.43	0.53	0.3
Neutral-N	0.378	0.358	0.39	0.47	0.4
Inversion-I	0.424	0.493	0.97	0.40	0.5

Attachment #2 to Thyroid Dose Table

Case #22 in the table corresponds to a calculation of the thyroid dose using all the AEC assumptions (presented in the AEC Safety Evaluation - Indian Point Unit No. 2, November 16, 1970 and Safety Guide 4, November 2, 1970) with the exception of X/Q values. Although similar meteorology is assumed by both Con Edison and the AEC, different formulation is used in the calculations. Con Edison uses the Sutton approach and the AEC uses the Pasquill method.

Adjusting the Con Edison values to AEC meteorological assumptions, thyroid doses of 195 rem (2 hours at the site boundary) and 267 rem (30 days at the low population zone) are obtained. These correspond to 180 rem (2 hours, SB) and 270 rem (30 days, LPZ) reported by the AEC in the Indian Point Unit No. 2 Safety Evaluation.

Question No.28 (J) (Tr.499) ASLB 1/19

"We would like to have a summary of some of the several monthly reports that have heretofore been submitted with reference to Indian Point No. 1, particularly as to releases of radioactive liquid and gases and compare those with the readings by the New York environmental surveillance groups and if there are any other surveillance groups...."

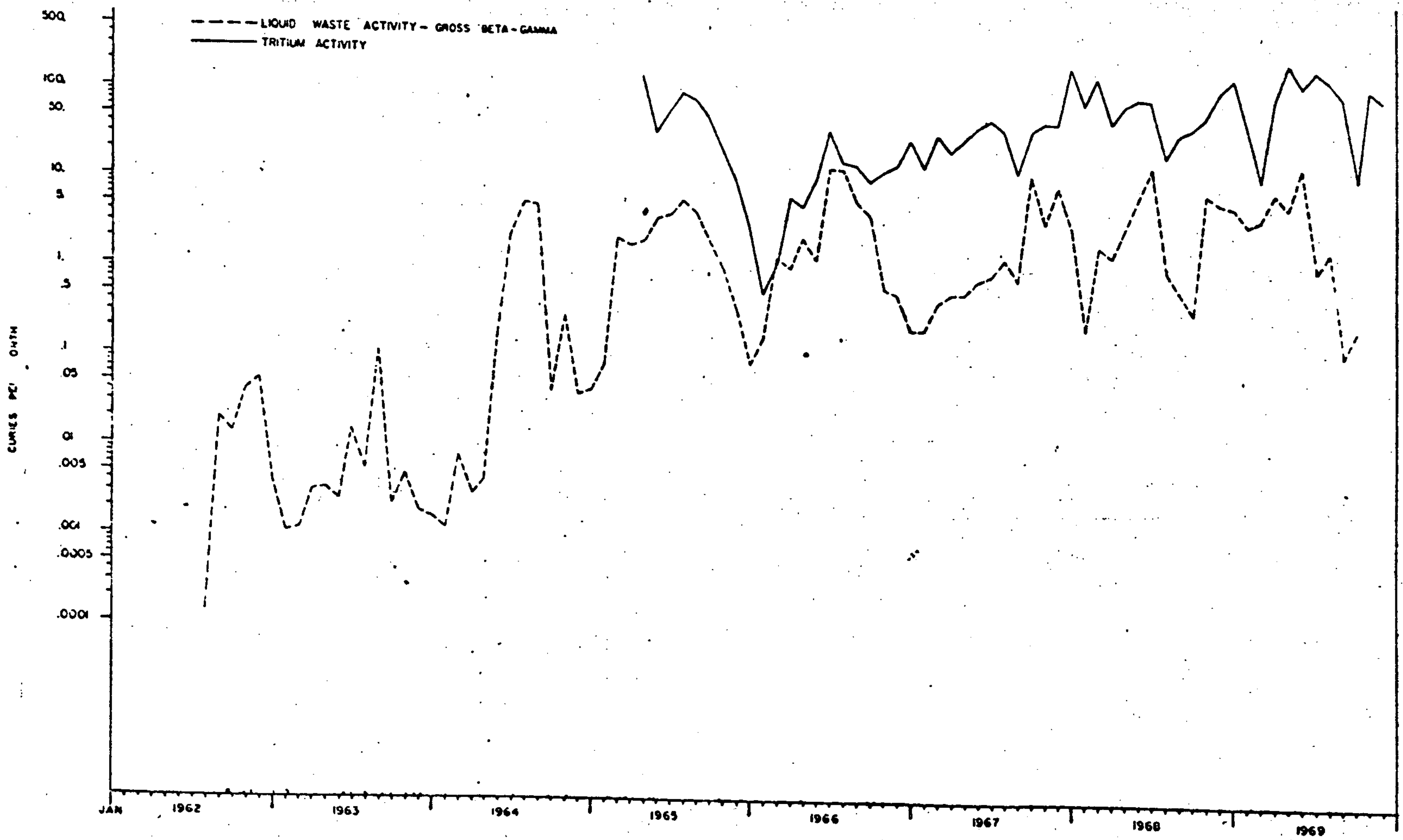
Answer

Two graphs are attached summarizing the liquid and gaseous releases from Indian Point Unit No. 1.

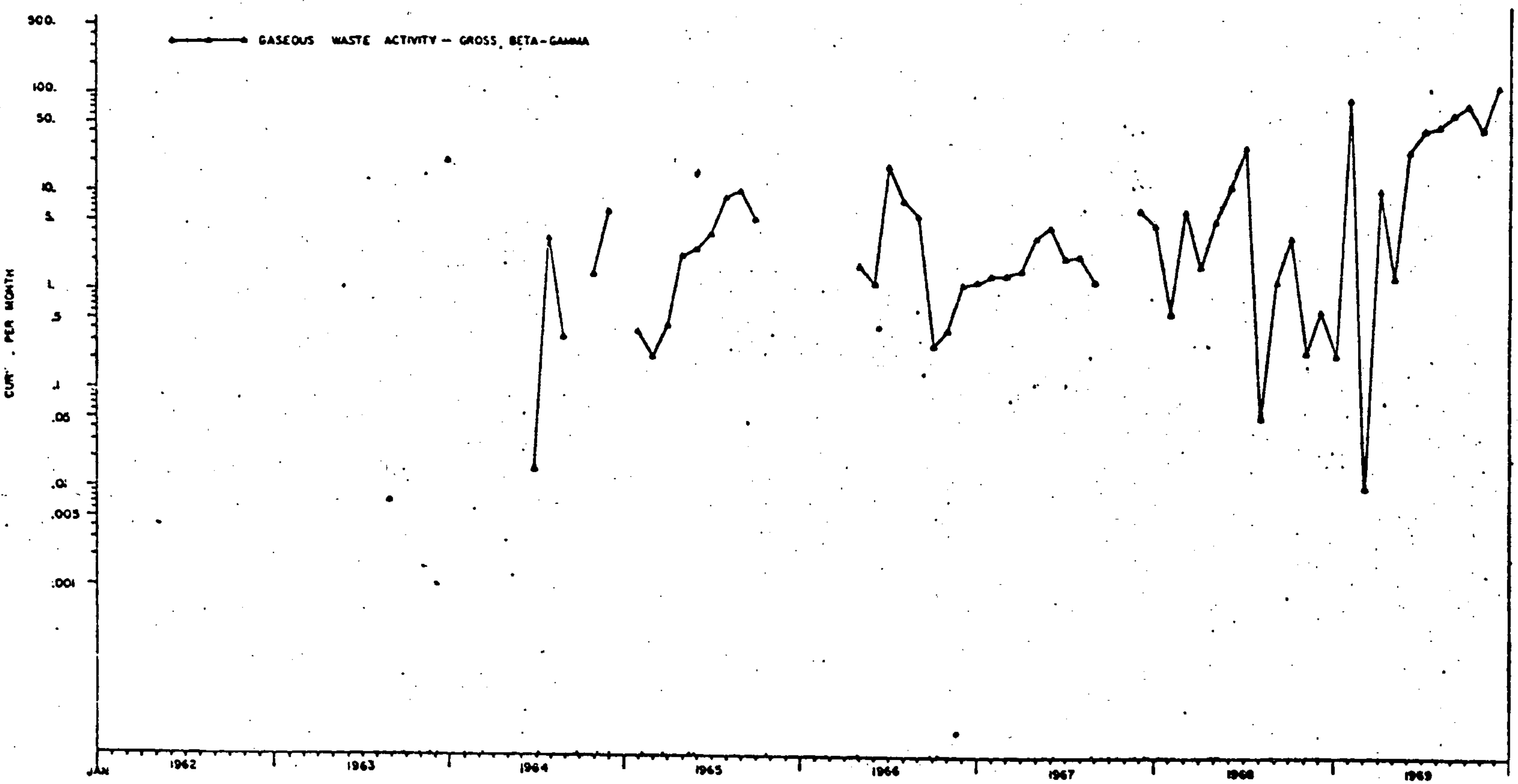
Also attached are Figures 1 - 17 which summarize the results of the Consolidated Edison environmental monitoring program. The dark vertical lines on these figures indicate the startup of Indian Point No. 1 in 1962 while the letters on the curves refer to rates which are given following the figures.

For the results of New York State environmental monitoring see the answer to Board question 26.

INDIAN POINT UNIT NO. 1
HISTORY OF LIQUID WASTE AND TRITIUM DISCHARGED



INDIAN POINT UNIT NO. 1
HISTORY OF GASEOUS WASTE DISCHARGED



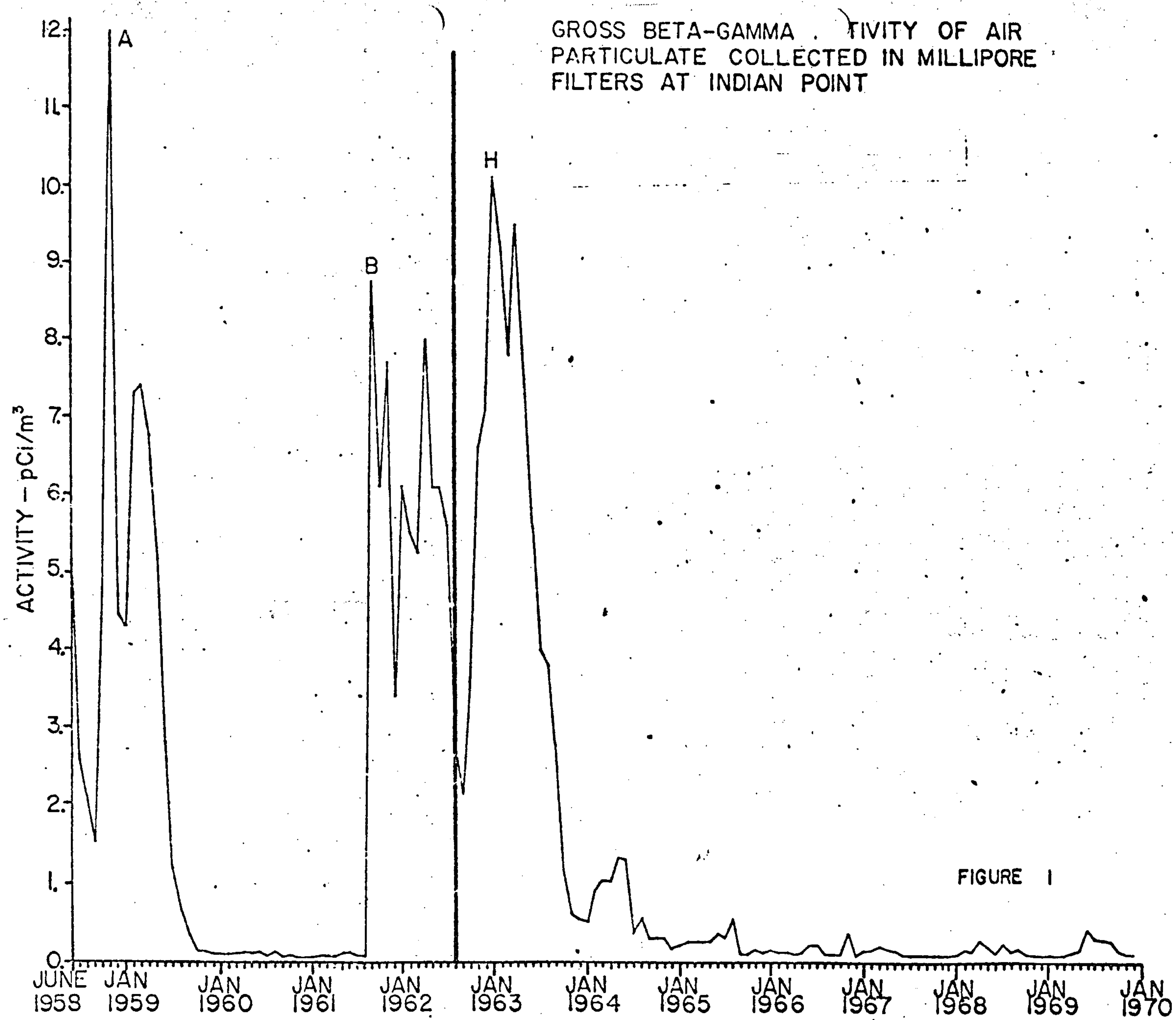

GROSS BETA-GAMMA . TIVITY OF AIR
PARTICULATE COLLECTED IN MILLIPORE
FILTERS AT INDIAN POINT
ACTIVITY – pCi/m³
A
B
H
0.
1.
2.
3.
4.
5.
6.
7.
8.
9.
10.
11.
12.
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966
JAN 1967
JAN 1968
JAN 1969
JAN 1970

FIGURE 1

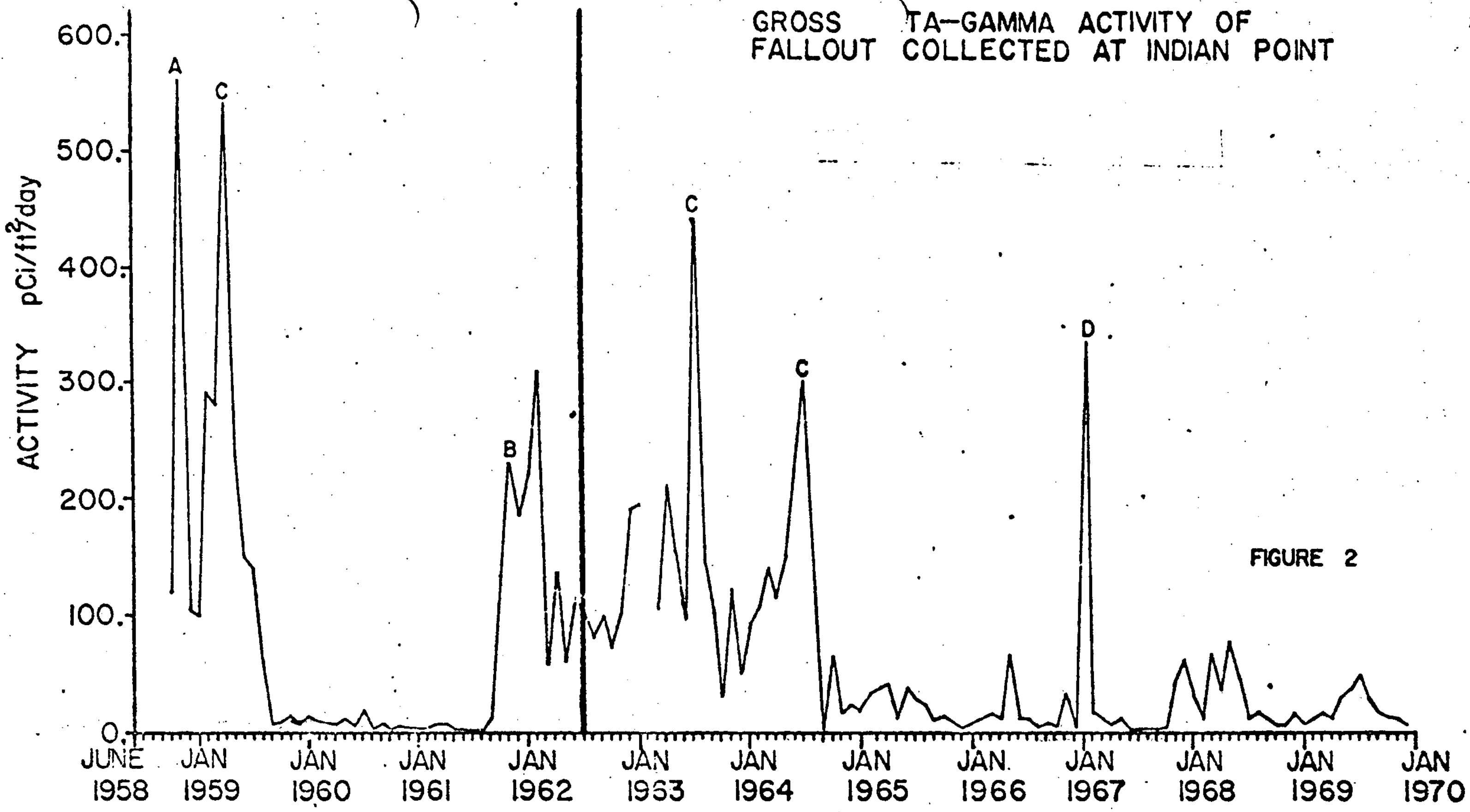

GROSS TA—GAMMA ACTIVITY OF
FALLOUT COLLECTED AT INDIAN POINT
ACTIVITY pCi/ft²/day
600.
500.
400.
300.
200.
100.
0.
A
B
C
D
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN. 1966
JAN 1967
JAN 1968
JAN 1969
JAN 1970

FIGURE 2

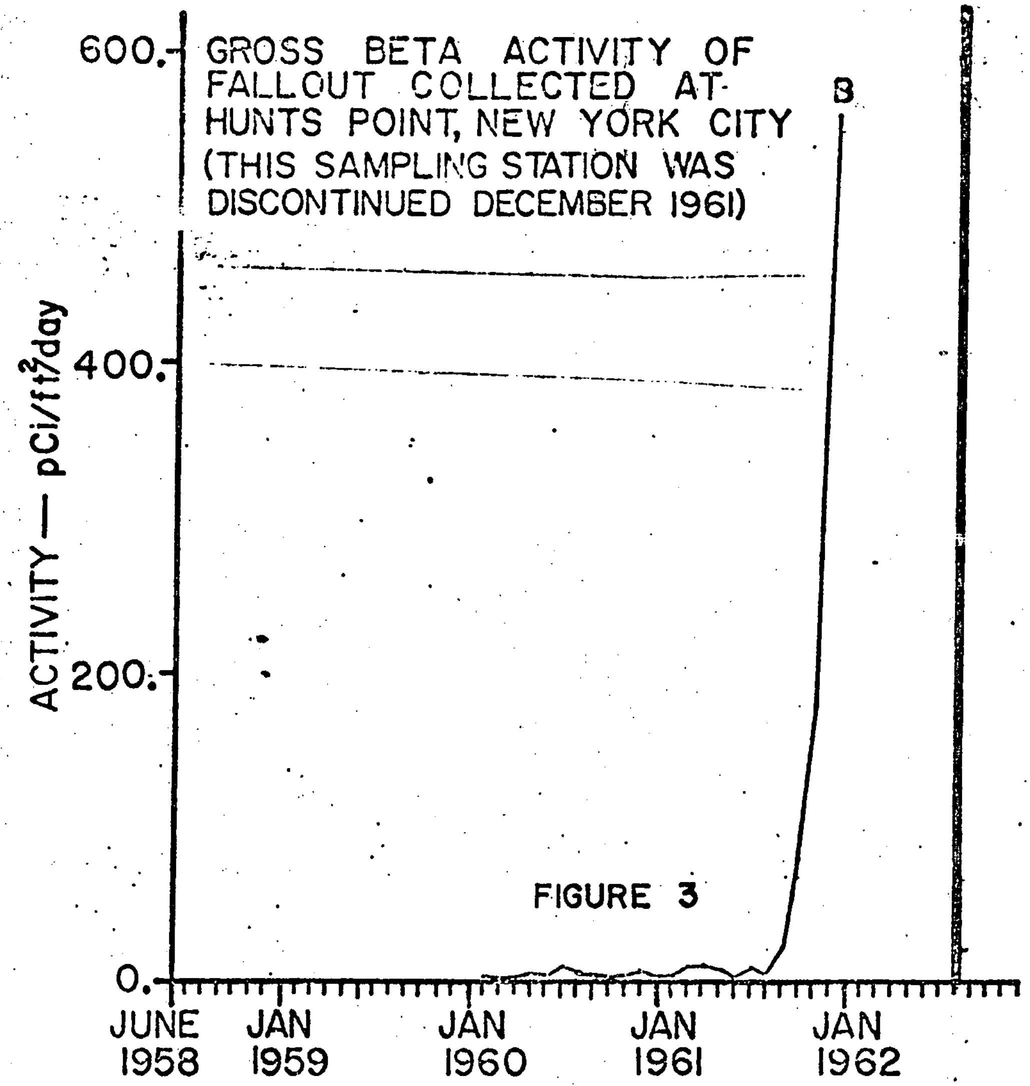

GROSS BETA ACTIVITY OF FALLOUT COLLECTED AT HUNTS POINT, NEW YORK CITY
(THIS SAMPLING STATION WAS DISCONTINUED DECEMBER 1961)
B
600.
400.
200.
0.
ACTIVITY — pCi/ft²/day
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962

FIGURE 3

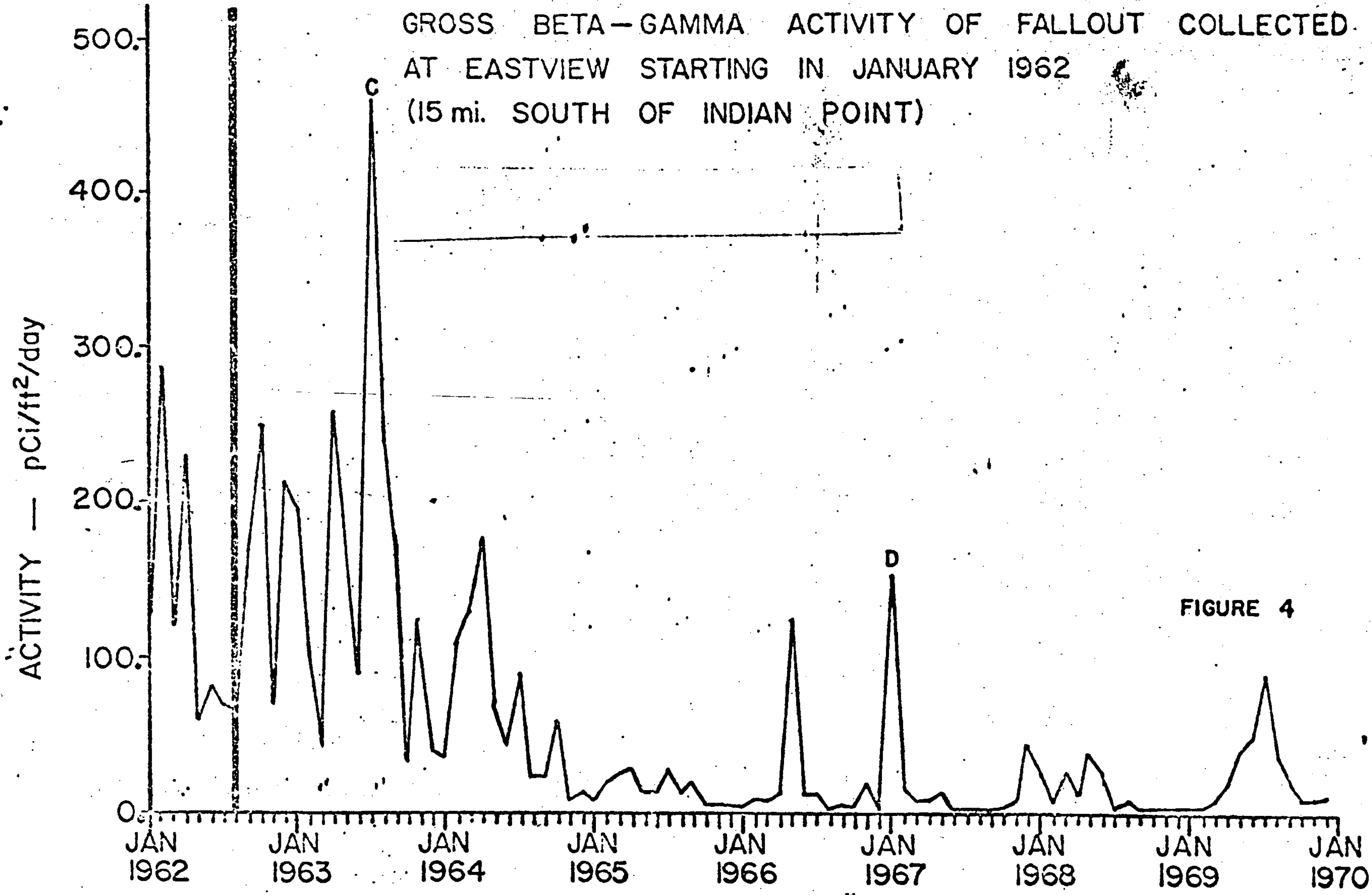

GROSS BETA-GAMMA ACTIVITY OF FALLOUT COLLECTED AT EASTVIEW STARTING IN JANUARY 1962 (15 mi. SOUTH OF INDIAN POINT)
ACTIVITY — pCi/ft²/day
500.
400.
300.
200.
100.
0.
B
C
D
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966
JAN 1967
JAN 1968
JAN 1969
JAN 1970

FIGURE 4

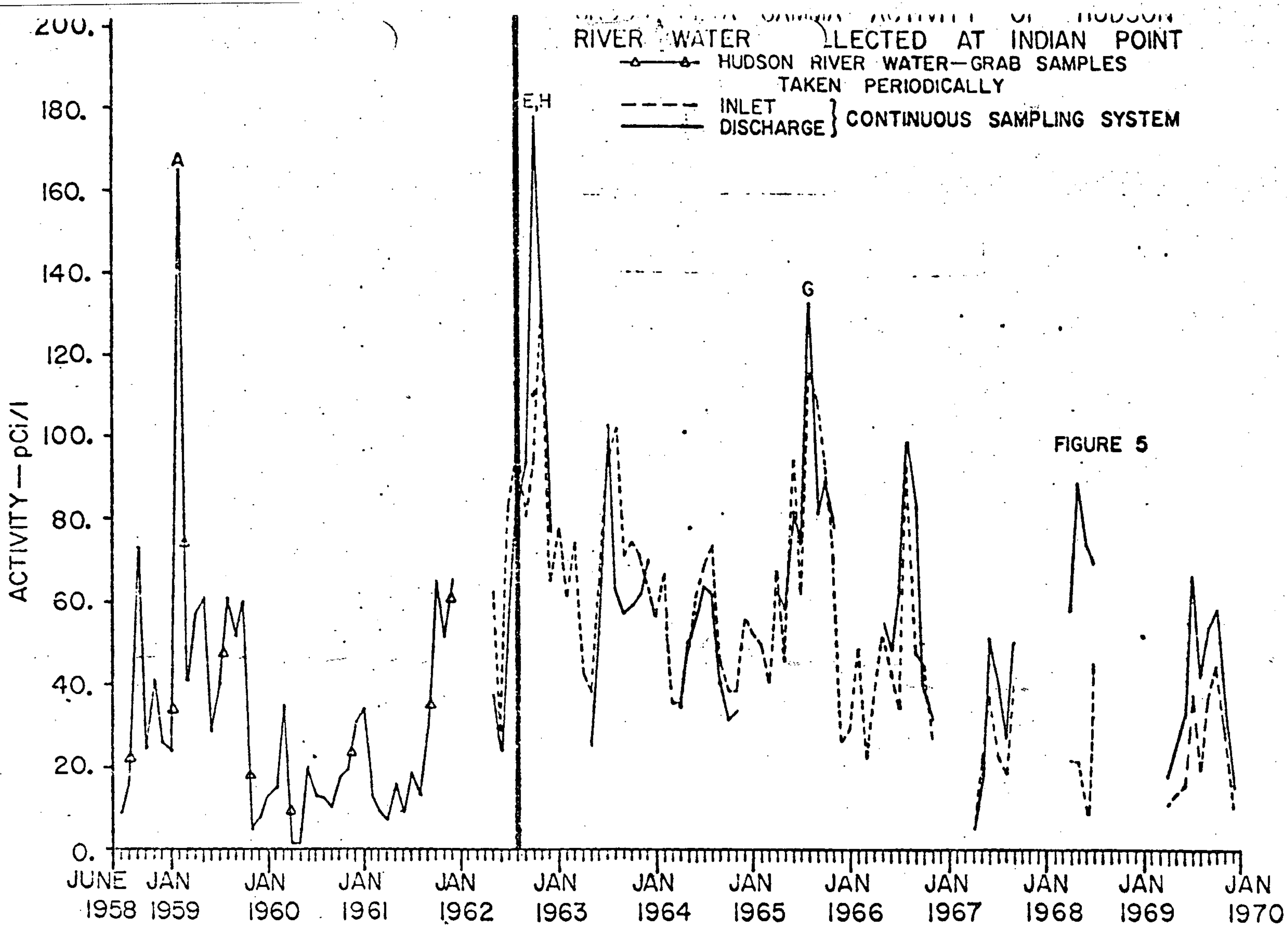

RIVER WATER LECTED AT INDIAN POINT
HUDSON RIVER WATER—GRAB SAMPLES TAKEN PERIODICALLY
INLET
DISCHARGE
CONTINUOUS SAMPLING SYSTEM
ACTIVITY—pCi/l
A
E,H
G
0.
20.
40.
60.
80.
100.
120.
140.
160.
180.
200.
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966
JAN 1967
JAN 1968
JAN 1969
JAN 1970

FIGURE 5

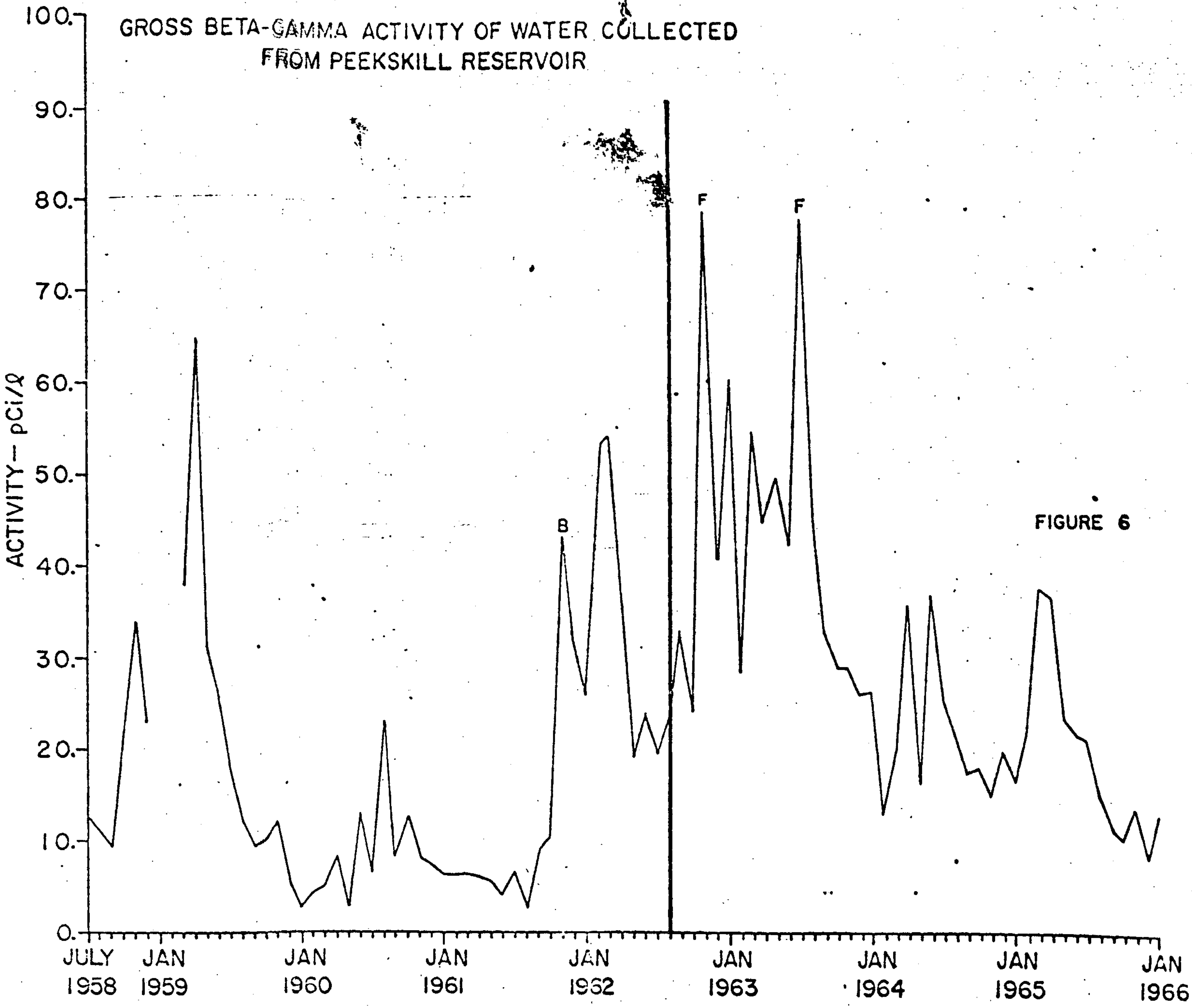

GROSS BETA-GAMMA ACTIVITY OF WATER COLLECTED
FROM PEEKSKILL RESERVOIR
ACTIVITY — pCi/ℓ
100.
90.
80.
70.
60.
50.
40.
30.
20.
10.
0.
B
F
F
JULY 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966

FIGURE 6

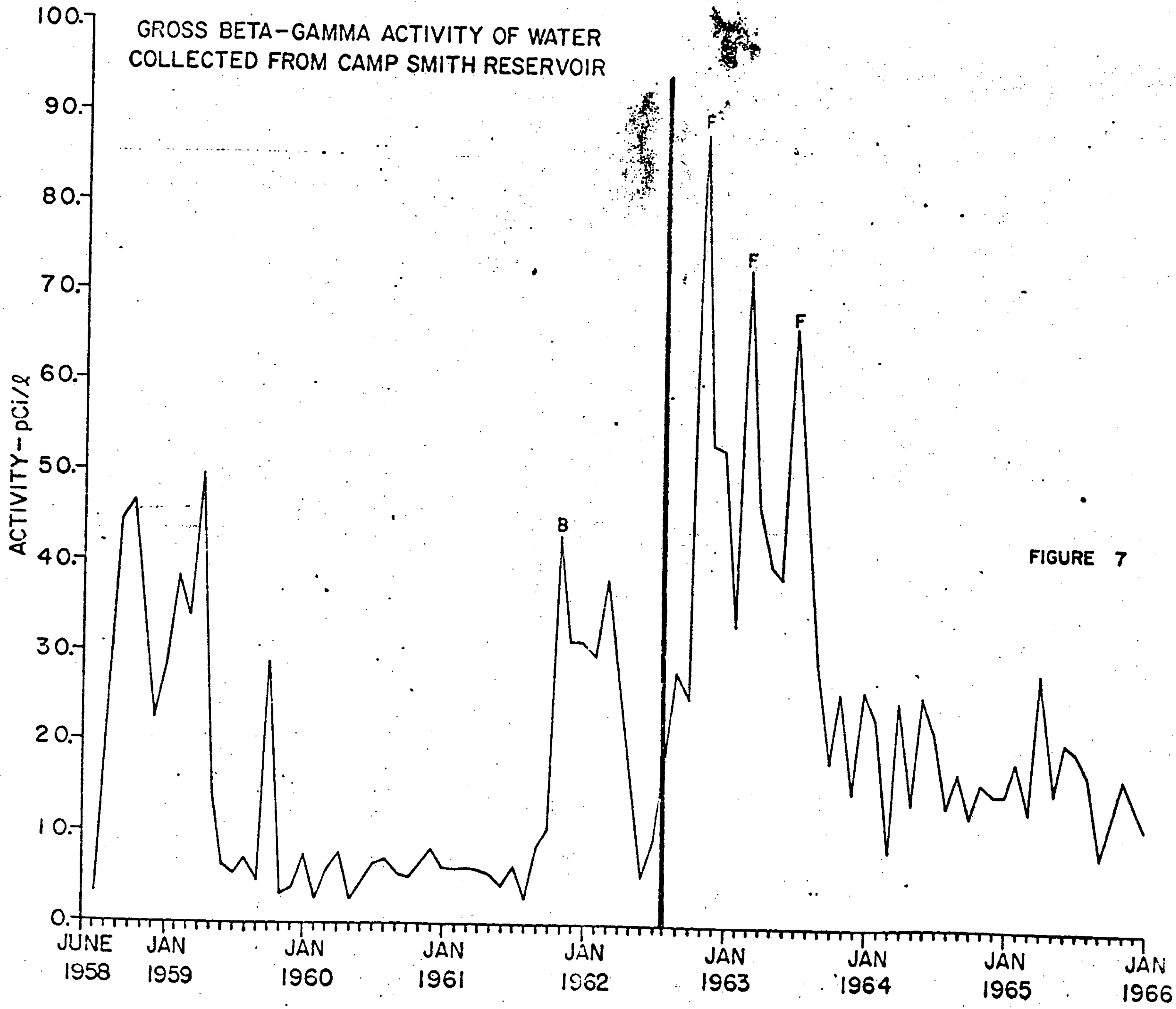

GROSS BETA-GAMMA ACTIVITY OF WATER
COLLECTED FROM CAMP SMITH RESERVOIR
ACTIVITY - pCi/ℓ
100.
90.
80.
70.
60.
50.
40.
30.
20.
10.
0.
B
F
F
F
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966

FIGURE 7

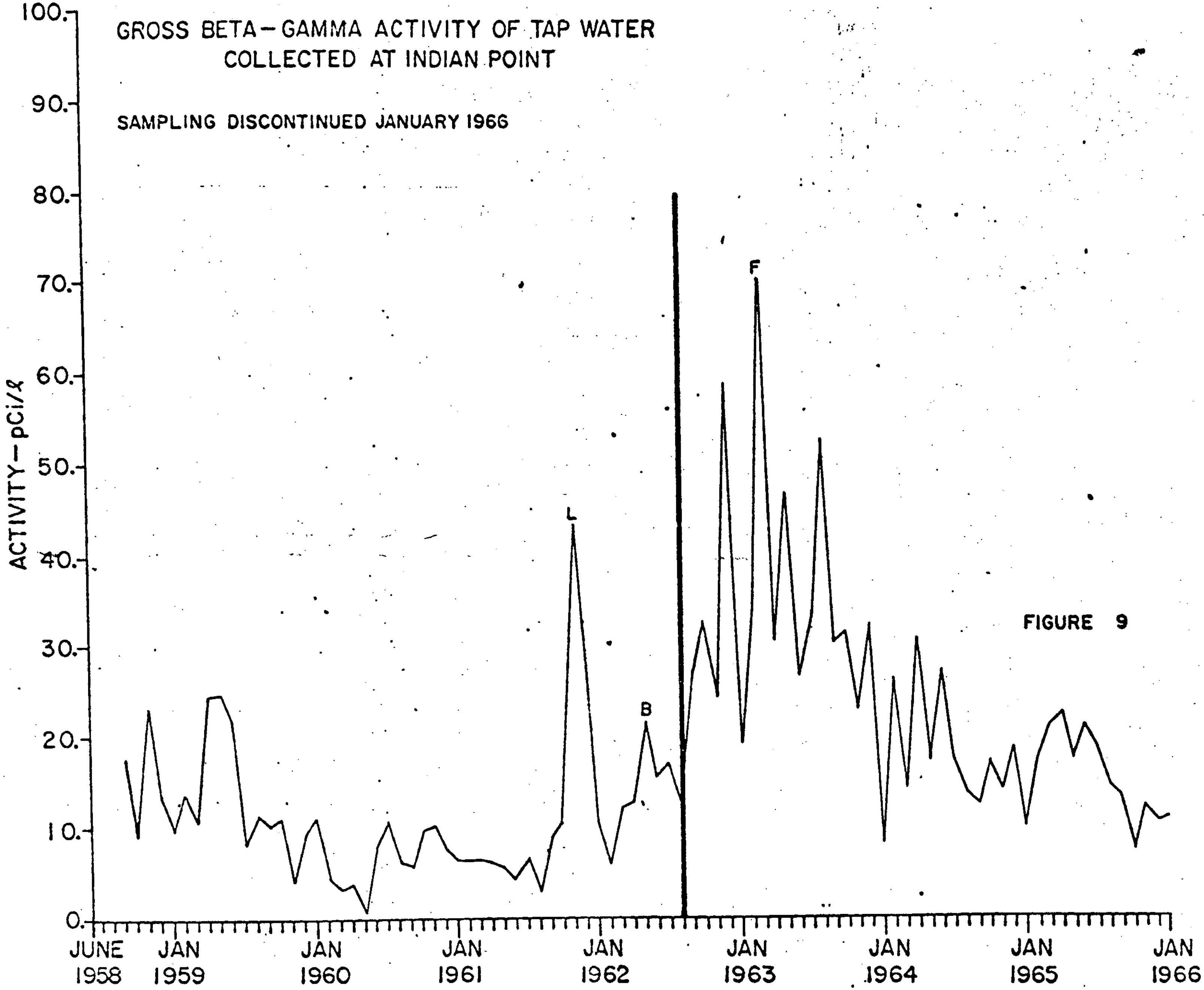
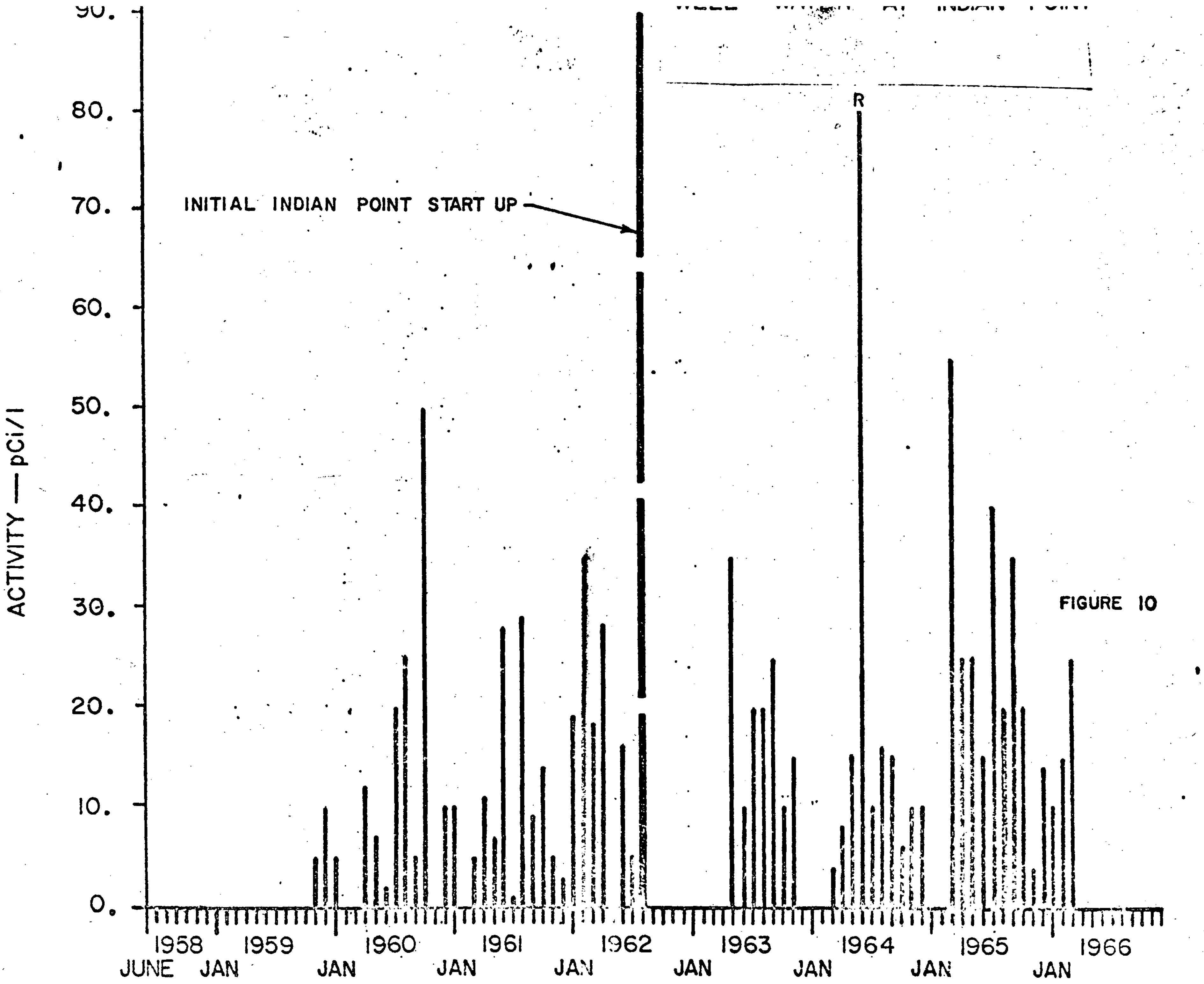

GROSS BETA-GAMMA ACTIVITY OF TAP WATER
COLLECTED AT INDIAN POINT
SAMPLING DISCONTINUED JANUARY 1966
ACTIVITY-pCi/ℓ
100.
90.
80.
70.
60.
50.
40.
30.
20.
10.
0.
L
B
F
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966

FIGURE 9


ACTIVITY — pCi/l
0.
10.
20.
30.
40.
50.
60.
70.
80.
90.
INITIAL INDIAN POINT START UP
R
FIGURE 10
1958
1959
1960
1961
1962
1963
1964
1965
1966
JUNE
JAN
JAN
JAN
JAN
JAN
JAN
JAN
JAN

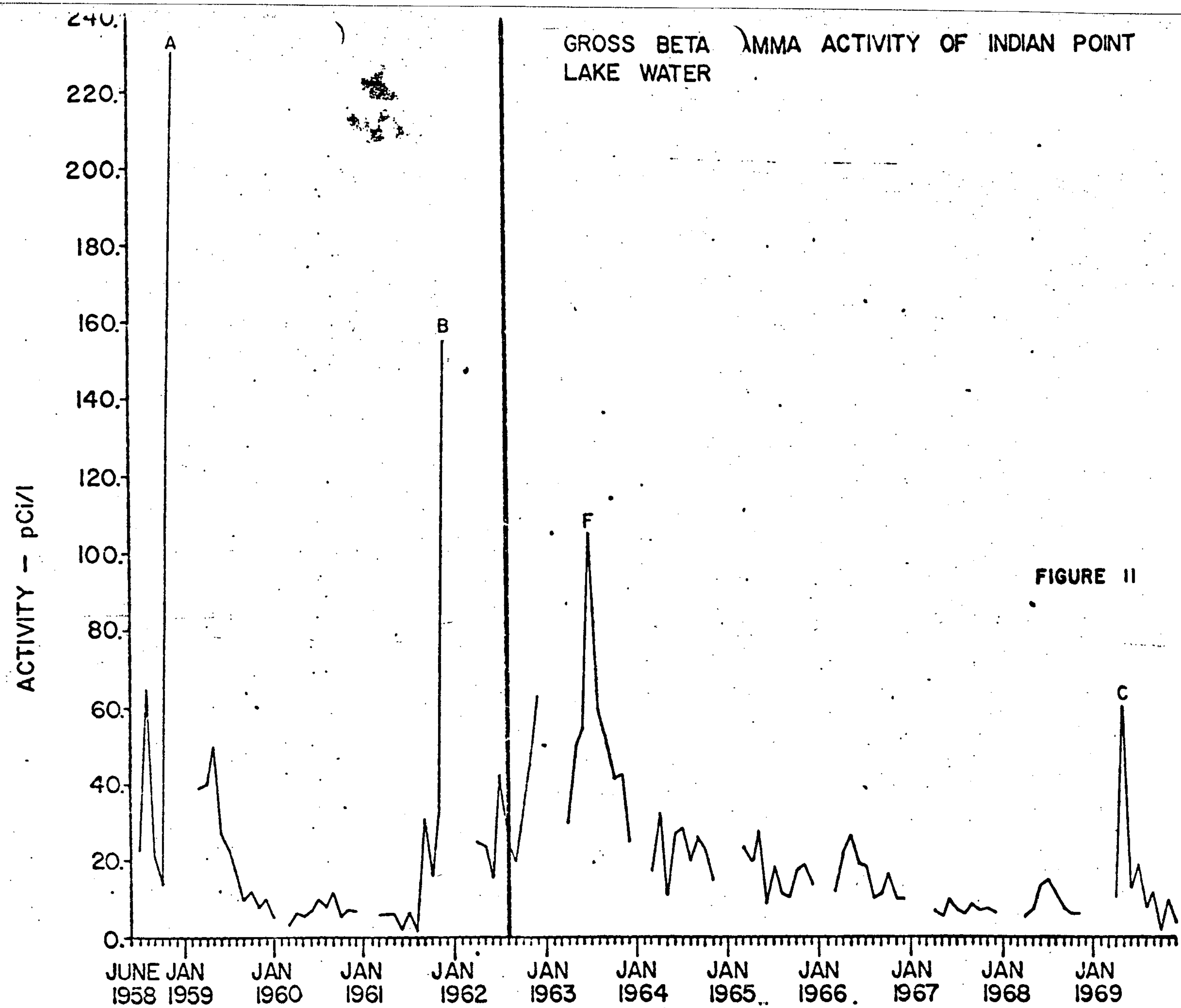

GROSS BETA)MMA ACTIVITY OF INDIAN POINT LAKE WATER
ACTIVITY – pCi/l
240.
220.
200.
180.
160.
140.
120.
100.
80.
60.
40.
20.
0.
A
B
F
C
JUNE 1958
JAN 1959
JAN 1960
JAN 1961
JAN 1962
JAN 1963
JAN 1964
JAN 1965
JAN 1966
JAN 1967
JAN 1968
JAN 1969

FIGURE 11

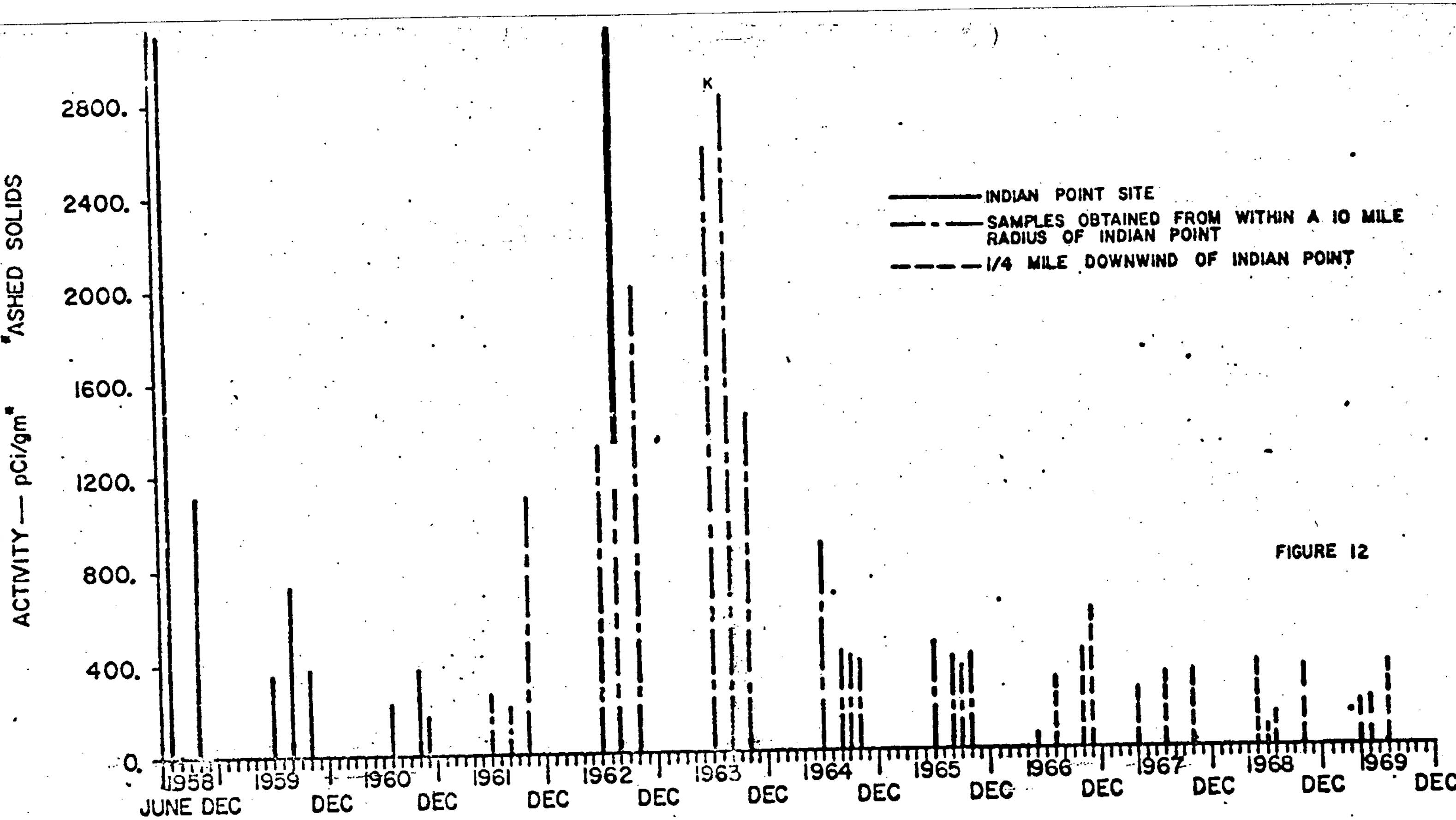
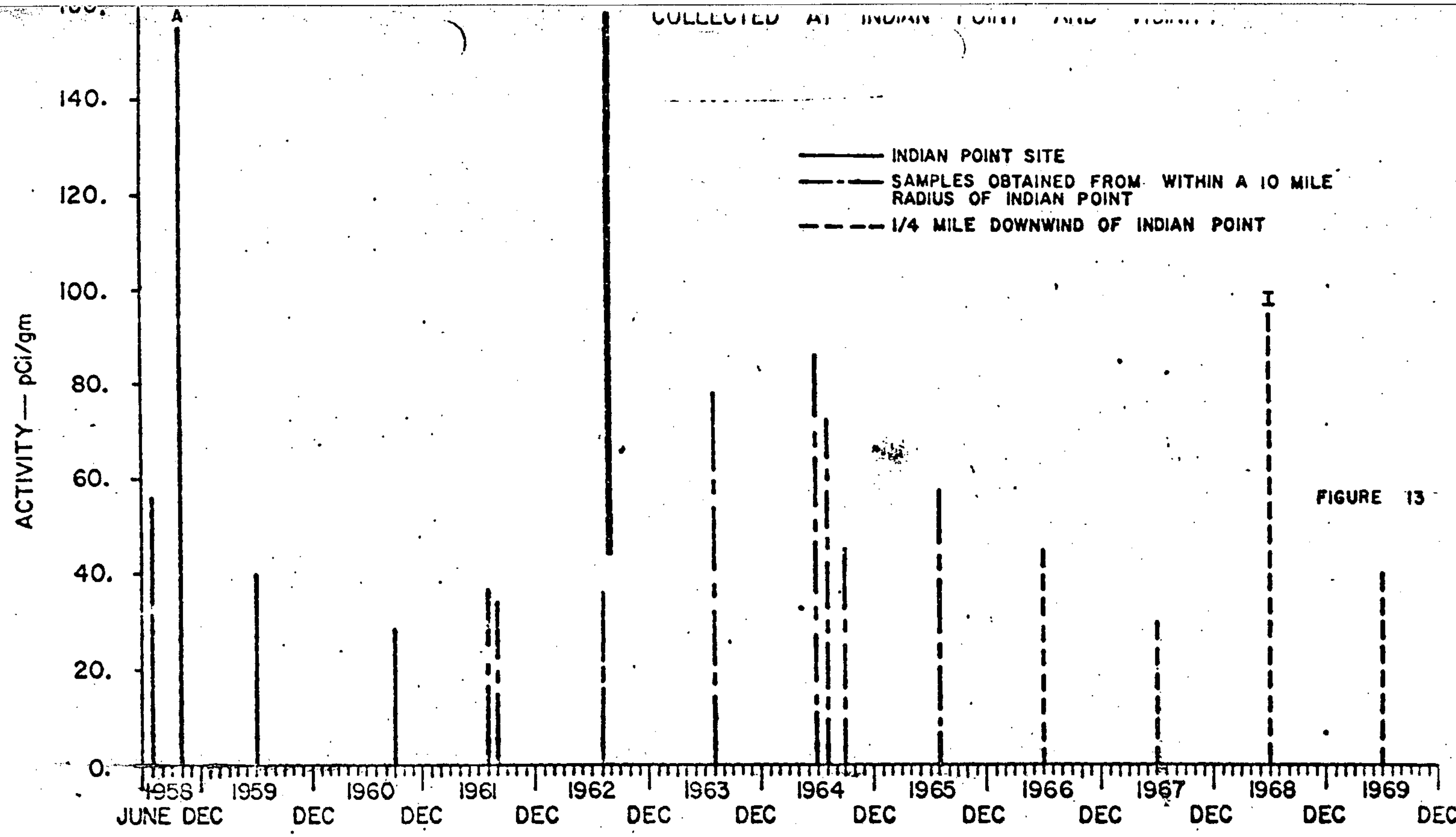

INDIAN POINT SITE
SAMPLES OBTAINED FROM WITHIN A 10 MILE RADIUS OF INDIAN POINT
1/4 MILE DOWNWIND OF INDIAN POINT
ACTIVITY — pCi/gm* *ASHED SOLIDS
2800.
2400.
2000.
1600.
1200.
800.
400.
0.
1958 1959 1960 1961 1962 1963 1964 1965 1966 1967 1968 1969
JUNE DEC DEC DEC DEC DEC DEC DEC DEC DEC DEC DEC DEC

FIGURE 12



INDIAN POINT SITE
SAMPLES OBTAINED FROM WITHIN A 10 MILE RADIUS OF INDIAN POINT
1/4 MILE DOWNWIND OF INDIAN POINT
ACTIVITY — pCi/gm
140.
120.
100.
80.
60.
40.
20.
0.
A
1958
1959
1960
1961
1962
1963
1964
1965
1966
1967
1968
1969
JUNE DEC
DEC
DEC
DEC
DEC
DEC
DEC
DEC
DEC
DEC
DEC
DEC
DEC

FIGURE 13

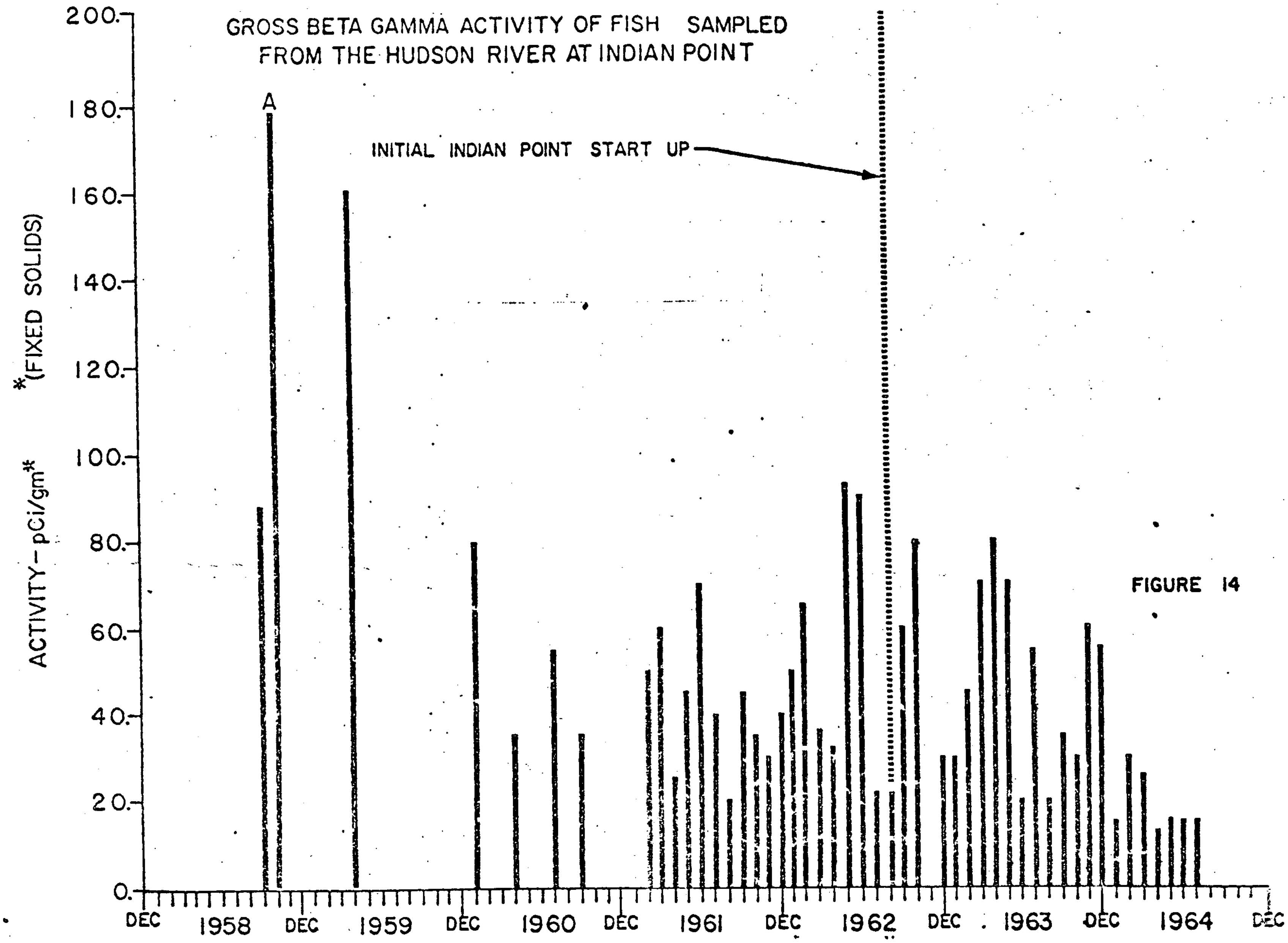

GROSS BETA GAMMA ACTIVITY OF FISH SAMPLED
FROM THE HUDSON RIVER AT INDIAN POINT
A
INITIAL INDIAN POINT START UP
ACTIVITY - pCi/gm*
*(FIXED SOLIDS)
200.
180.
160.
140.
120.
100.
80.
60.
40.
20.
0.
DEC
1958
DEC
1959
DEC
1960
DEC
1961
DEC
1962
DEC
1963
DEC
1964
DEC

FIGURE 14

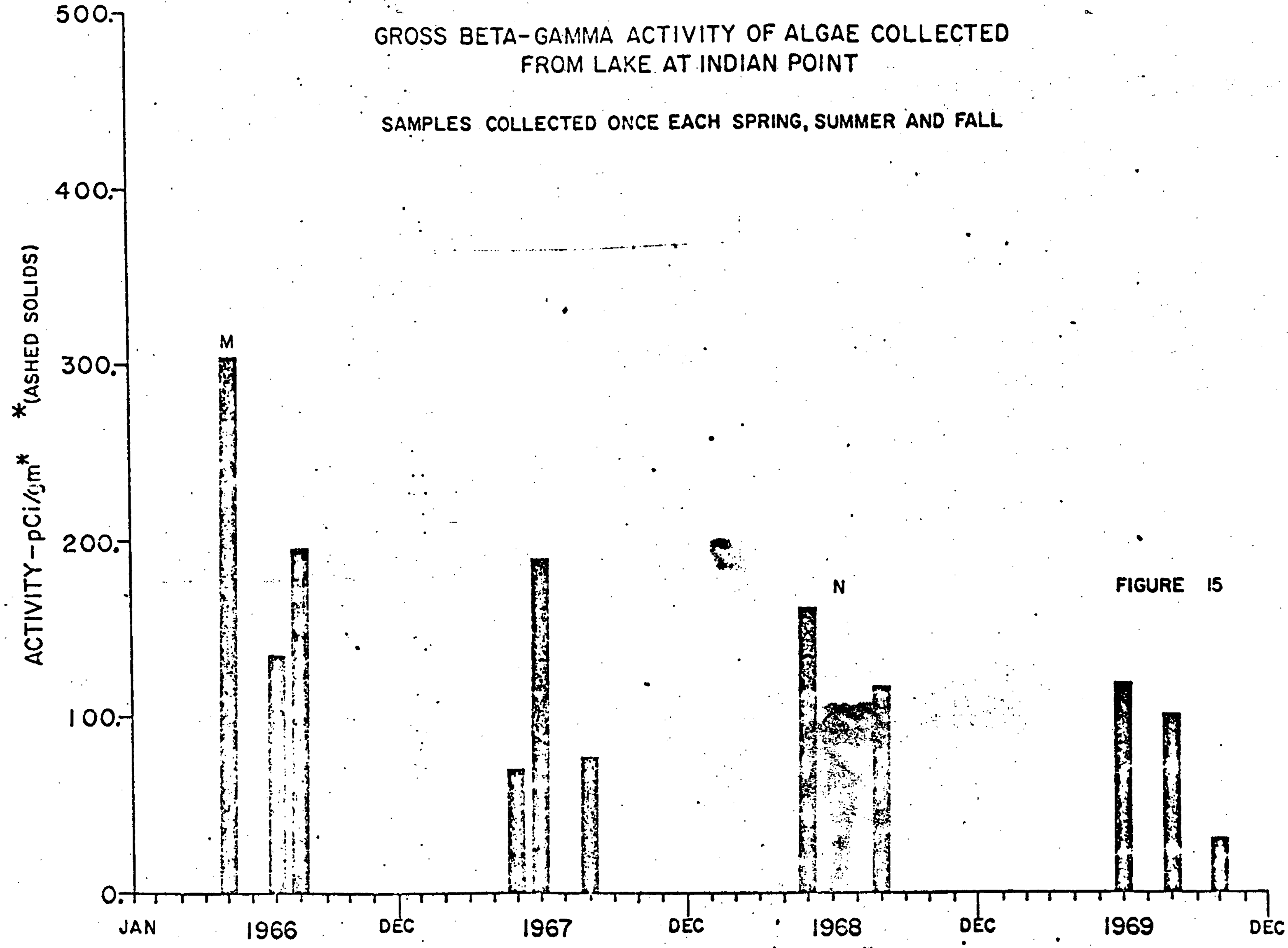

GROSS BETA-GAMMA ACTIVITY OF ALGAE COLLECTED
FROM LAKE AT INDIAN POINT
SAMPLES COLLECTED ONCE EACH SPRING, SUMMER AND FALL
ACTIVITY-pCi/gm*
*(ASHED SOLIDS)
500.
400.
300.
200.
100.
0.
M
N
JAN
1966
DEC
1967
DEC
1968
DEC
1969
DEC

FIGURE 15

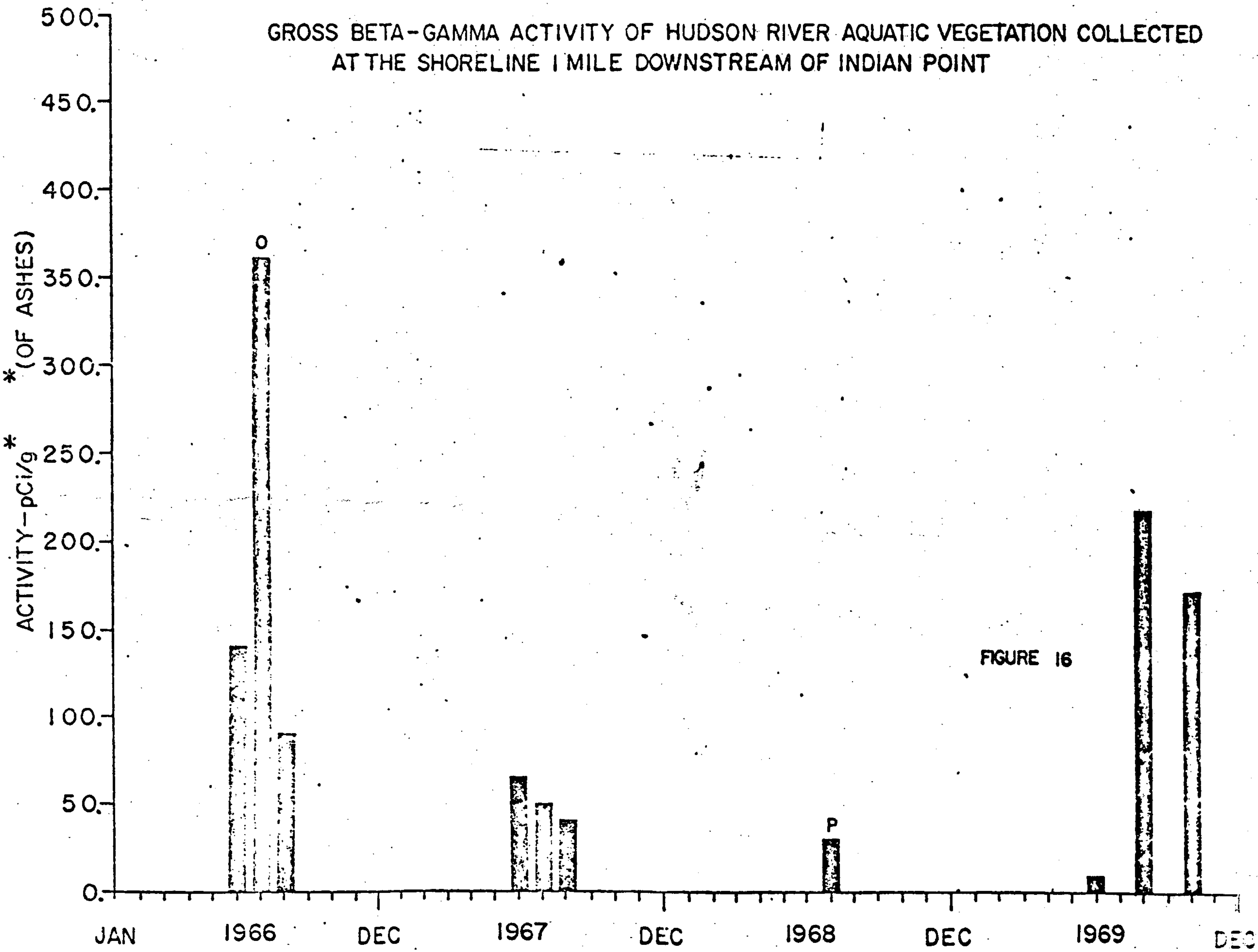
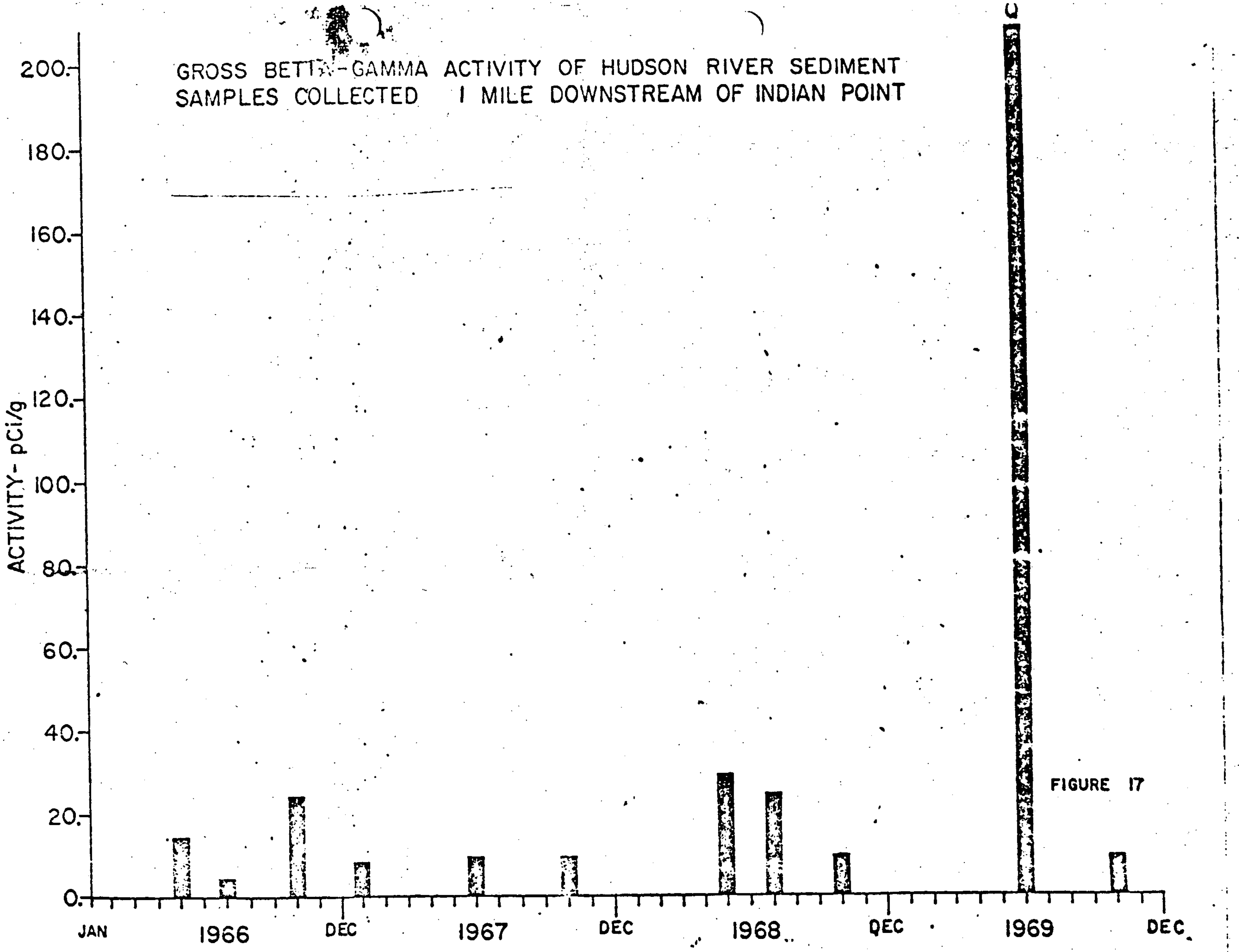

GROSS BETA-GAMMA ACTIVITY OF HUDSON RIVER AQUATIC VEGETATION COLLECTED
AT THE SHORELINE 1 MILE DOWNSTREAM OF INDIAN POINT
ACTIVITY-pCi/g* *(OF ASHES)
500.
450.
400.
350.
300.
250.
200.
150.
100.
50.
0.
O
P
JAN
1966
DEC
1967
DEC
1968
DEC
1969

FIGURE 16



GROSS BETA-GAMMA ACTIVITY OF HUDSON RIVER SEDIMENT
SAMPLES COLLECTED 1 MILE DOWNSTREAM OF INDIAN POINT
ACTIVITY- pCi/g
200.
180.
160.
140.
120.
100.
80.
60.
40.
20.
0.
JAN
1966
DEC
1967
DEC
1968
DEC
1969
DEC

FIGURE 17

The Following Notes Pertain To Figures 1 Through 17.

(A). Intensive atmospheric nuclear weapons testing by both the United States and Russia in October 1958. Fallout from these tests was reflected in marked increases in the gross beta activity of air, water, soil and vegetation samples.

(B). Fallout resulting from the resumption of atmospheric weapons testing reversed a declining trend in the amount of background radioactivity. This is readily apparent from the results of measurements on media collected after September 1961 and is in agreement with measurements made by other agencies in this geographic location.

(C). Increase in activity (fallout) attributed to the transfer of aged radionuclides (due to high yield atmospheric weapons testing) from the stratosphere to the troposphere.

(D). Spectral analysis of the fallout samples showed predominantly fresh fission products. This increase is attributed to the Chinese atmospheric nuclear weapons testing in October and December 1966 and the increase in rainfall just prior to the January 1967 collection period.

(E). The salinity of the Hudson River increases progressively from the spring to fall of each year. This salt front pushes its way upriver, thus salinity content increases seasonally.

(F). An examination of measurements taken in the period February 1, 1963 through July 31, 1963 indicates that atmospheric fallout is still the dominating influence in most samples. In addition the samples from non-flowing surface water sources are still increasing in radioactivity from the accumulation of the longer lived fallout radionuclides.

(G). Values of Potassium 40 found in the Hudson River water samples increased in the draught years 1964-1966 when low precipitation conditions increased the sea water intrusion. Thus giving a higher activity.

(H). In 1962, and early 1963 air particulate, fallout and water samples showed higher average values than those obtained in 1961.

(I). This slightly higher activity reflects the increase due to the Chinese atmospheric nuclear weapons testing of December 1967.

(J). Effects of atmospheric nuclear weapons testing.

(K). Intermediate lived fresh fission products characteristic of weapons testing fallout such as Niobium-95, Ruthinium-106 and 103, Zirconium-95 and Cerium-141 were found in the 1963 vegetation samples. In addition, meteorological conditions for the latter half of 1963 was characterized by an unusual lack of precipitation.

(L). Samples of drinking water in 1961 were obtained from locations within a ten-mile radius of Indian Point. The following is a breakdown of that years drinking water data:

NO. OF SAMPLES	GROSS BETA ACTIVITY pCi/l		
	Minimum	Maximum	Average
209	less than 1	286	10

(M). The May 1966 samples of algae collected from the Indian Point lake was only analyzed for Iodine-131 and none was detected. A complete spectrum analysis to detect the presence of other isotopes was not performed unitl the July samples were taken. The July samples showed predominately fresh fission products characteristic of weapons testing. In addition, the May fallout samples taken at Indian Point and Eastview showed an increase of gross beta gamma levels of approximately eight times the average.

(N). Only two samples reported in 1958.

(O). Algae is known to be a concentrator of radioactive isotopes, such as Iodine, Cobalt and Manganese. The samples of green slime scraped from Hudson River shore rocks collected at three sampling points, up to 2 miles downstream of the discharge canal, indicate the presence of Co58, Co60 and Mn54 in slightly higher concentrations than may be expected from fallout or other sources.

(P). Dredging operations in connection with the construction of Units 2 and 3 have affected algae growth to the extent that only one month (July 1968) sample was collected and analyzed.

(Q). Samples of bottom sediment were collected in the discharge canal and at four locations near the shoreline at various distances downstream of the plant. These were measured for gross beta radioactivity and a qualitative analysis made to determine radionuclide content. The gross beta radioactivity of these samples for 1968 is higher than the levels reported in the previous period and several are beyond the range of levels found in recent years which ranged from 10 to 120 picocuries per gram. Manganese-54 and Cobalt-60 can be attributed to plant releases while Potassium-40 is due to the natural salinity of the water and Cs 137 partly due to nuclear weapons testing and partly attributable to plant releases.

(R). Of the ten well water samples collected once each month, from Indian Point in 1964, the following data was tabulated:

Collection Month	GROSS BETA - GAMMA ACTIVITY pCi/1 Suspended Solids	Dissolved Solids	Total Activity
March	2 $\pm$ 3	2 $\pm$ 4	4 $\pm$ 5
April	3 $\pm$ 3	5 $\pm$ 4	8 $\pm$ 5
May	10 $\pm$ 4	5 $\pm$ 4	15 $\pm$ 6
June	50 $\pm$ 3	30 $\pm$ 4	80 $\pm$ 5
July	5 $\pm$ 3	5 $\pm$ 4	10 $\pm$ 5
August	10	6	16
September	5	10	15
October	3	3	6
November	5	5	5
December	5	5	10

Precipitation data (1) at four locations within a fifteen mile radius of Indian Point indicates that the summer months of 1964 were ones of severe draught conditions.

(S). This one isolated area, approximately 2 miles South of the Bear Mountain Bridge along Rout 9W is more than 10 times higher than other nearby sampling points. This level has remained consistantly high since readings were first taken and is believed attributable to a vein of uranium ore.

(1) Data obtained from U.S. Department of Commerce, Weather Bureau Climatological data for New York Annual Summary 1964, Volume 76, No. 13.

29. (J) (Tr. 500) ASLB 1/19

"Now, the applicant environmental impact statement in Appendix D stated on Page 2, thereof, if the average release rate from the plant vent is greater than 10 percent of the annual allowable release rate as specified in Paragraph 3.9-(B)1 during the month just ended, an environmental survey shall be conducted in accordance with 3 for the subsequent months.

"I couldn't find Paragraph 3.9-Cl and if that could be submitted, I would be happy to have it with the figures that are available."

<u>ANSWER</u>:

The Paragraph 3.9-Cl referred to is in the Applicant's proposed Technical Specifications, Page 3.9-2, Paragraph 3.9.C, Gaseous Effluents, Item 1.

30. (B) (Tr. 500)

"In the design of the plant you mention that the ECCS system was, according to reports, made more reliable and this permitted the removal of the crucible below the reactor and other considerations did too, apparently.

I would like to reemphasize the need for discussion of the research and development results that have led to the conclusion of the very high reliability that is attributed to the ECCS system."

ANSWER

For details of R&D related to emergency core cooling, see response to Board Question No. 10. Further information relative to this question will be contained in the forthcoming response to Board question No. 16.

31. (B) (Tr. 500)

"In the report there is indicated that certain changes or conditions will be required such as purging the containment or removal of the hydrogen, adding filters to the ventilation system.

I would like to have an indication as to why these changes or additions are not required before the plant goes into operation, why it is possible to let some changes or additions come along a year or two or three years after the plant begins to operate.

What considerations led to the conclusion that these could be delayed?"

Answer

AEC staff response

"As I read the reports the plant was not originally designed on the basis of taking into consideration the design basis formally. Calculations have been made to show what some of the resistance of some of the structures would be. I would like to have some discussion of what effects could be expected and, if you wish, what the probability would be of the design basis tornado interacting with the control room, the building in which the control room is located and also the building in which the decelerators are located and the effect that one could expect on the source of emergency power."

Answer:

Indian Point Unit No. 2 does not have a design basis tornado criterion. The capability of Indian Point Unit No. 2 to withstand high winds is stated in the answer to Question 1.11 of the FSAR.

The nearest weather stations to Peekskill having wind recording instruments capable of recording wind gusts of 100 mph or greater, and with records for twenty years or more, are the following:

Newburgh, N.Y. / Steward Air Force Base
Bedford Mass. / L. G. Hanscom Field
Atlantic City, N.J. / Naval Air Station, and Weather Bureau
Rome, N.Y. / Griffiss Air Force Base
Patuxent River, Md. / Naval Air Station

The National Weather Records Center, Asheville, N.C. searched the records of the above stations for observations of wind speeds greater than or equal to 100 mph. Only two cases of 110 mph and 111 mph maximum gusts, were found exceeding 100 mph. Both these cases occurred during the passage of hurricanes.

Question No. 33 (B) (Tr. 501)

"There is a statement in the staff Safety Evaluation that on the basis of the very low probability for wind speeds greater than 100 miles an hour at the Indian Point site and the resistance of these structures, that the unit is adequately protected against by winds.

I may have missed in the records any history of wind speeds greater than 100 miles an hour in this general area. If I have, I would like for someone to call to my attention the place where this reference is located. If not, is there information available on the frequency, the number of times when winds in this general area have exceeded 100 miles an hour."

Answer

See answer to Board Question No. 32.

Question No. 34 (B) (Tr. 502) ASLB 1/19

"On page 36 of the staff Safety Evaluation it is indicated that the Indian Point 2 reactor vessel cavity is designed to protect the containment against missiles that might be produced by postulated failure of the reactor vessel and it goes on to discuss some of this protection. The question here is concerned with whether the emergency core cooling system and the other provisions that have been made take into account such failure and, if not, why not?

Answer

The design bases of the ECCS does not take into account a postulated rupture of the reactor vessel because rupture of the reactor vessel is not considered credible. The reactor vessel is conservatively designed and carefully constructed with strict attention to quality control and quality assurance. Reactor operating limits and a responsible in-service inspection program are established by the Technical Specifications, which assure safe operation. Together, these eliminate the probability of reactor vessel rupture. The cavity design features referred to were incorporated on the recommendation of the ACRS at the time of its Construction Permit review.

35. (B) (Tr. 502)

"In several places it is indicated that the applicant has provided results of analyses which indicate that the consequences of failure to scram during transients are tolerable for the existing Indian Point unit to desire at a power level of 2358 megawatt thermal. It says additional studies are required for this general question.

I would like to know what additional study is being made, where there are results of such study and what the schedule is for completing those studies?"

ANSWER

Studies have been performed in addition to those determining the consequences of failure to trip. These additional studies involved a detailed failure analysis, using as a representative Westinghouse system the Indian Point Unit 2 reactor protection system, considering both random component failures and systematic or common mode failures. The purpose was to assess the likelihood of failure to trip during anticipated transients to determine whether it is acceptably small.

A probabilistic analysis of trip failure was performed considering random component failure as well as a detailed qualitative study of common mode failures which could prevent trip. Measures taken in design, construction, operation and maintenance to minimize common mode failures were also evaluated. Results indicate a very remote probability of failure to trip (2×10^{-7}/demand) due to random component failure. The detailed evaluation of potential common mode failure also showed that adequate preventative measures have been undertaken such that the likelihood of failure to trip is acceptably small. The details of this study will be presented in a Westinghouse report to be submitted to the AEC later this month.

Question No. 36 (J) (Tr. 502)

"I have an Appendix C to the Safety Evaluation by the Staff. It bears the number 900 but it looks to be a portion of a letter from the Air Resources Environmental Laboratory. It seems like it should be followed by another letter but I do not have it. If that could be supplied or I assume it is an error in the assembly, that part of that page is missing. But the page that I do have, however, raises some matters and your attention is directed to the entire item.

But the last sentence of the first paragraph says in reference to the original documentation of the Indian Point site about winds within certain sectors and so forth and says "Although this point is at a distance 580 meters from Unit 2, it is not in the most prevalent wind direction by a considerable amount."

Answer

The statement from the Air Resources Environmental Laboratory applies in general to applicant's method of calculating the average annual dilution factor (χ/Q) which will be applied to determine the release rate for gaseous effluents from the site.

The suggestion is that χ/Q be calculated in the sector with the most prevalent wind direction. The distance to the site boundary in the sector with the most prevalent wind is greater than 580 meters (823 meters). Applicant has calculated χ/Q in this and several other sectors, and has found that the most restrictive limit

(χ/Q) is not in the sector with the most prevalent wind direction. The χ/Q value as presented in Applicant's proposed Technical Specification was a result of calculation with the worst combination of sector meteorology and distance to the site boundary and therefore is more conservative than that which was proposed by the AREL.

Question No. 37 (J) (Tr. 503) ASLB 1/19/71

"Air Resources Environmental Laboratory state in their third paragraph: It is our view that the use of the building wake effect in the long-term average diffusion equation, as was done by the applicant, is inappropriate.

"Was there a further computation made by eliminating the building wake effect and, if so, what results derived from that computation?"

Answer

Yes, a further computation was made eliminating the building wake effect. Applicant computed a value of X/Q, the average annual dilution factor for Indian Point Unit No. 2 of 2.05×10^{-5} sec/m^3 without wake and suggested this for the proposed Technical Specifications. After discussions with the AEC staff, an even more conservative value for X/Q of 2.5×10^{-5} sec/m^3 was agreed upon for the revised proposed Technical Specifications, Section 3.9.C.1.

38. (J) (Tr. 503)

"The last preceding sentence of the second paragraph says "The only explanation we have for the ESSA Value"-- and I take it that is the Environmental Science Services Administration--" being twice as high is the use of the building wake effect in the Applicant's assumptions.

"So I wonder if that matter could be either recalculated or reconsidered and comments of both the staff and the Applicant given in that regard?"

ANSWER

Although Applicant believes that the use of the building wake effect assumption is reasonable, the value of X/Q in the revised proposed Technical Specifications was calculated without this building wake effect. (See response to ASLB Question 37).

39. (J) (Tr. 504)

"We would like to have also a comparison between the R&D indicated to be necessary at the construction permit stage at Indian Point No. 2 and that which is indicated or advisable at the operating stage of Indian Point No. 2.

Why have there been changes and what data has been developed to indicate that others are indeed advisable? We call your particular attention to the findings submitted by both the staff and the applicant in that regard as well as the Board's decision which was issued at the time of the construction permit for Indian Point No. 2."

ANSWER

Only one R&D program has been added to the list of items listed as necessary for plant operation since the construction permit stage of Indian Point #2. That program is the Containment Spray Program. Based upon work done at ORNL and BNWL, the iodine removal aspects of the spray and spray additive have been studied experimentally and analytically.

In addition, the Containment Air Recirculation Filter studies, required at the construction permit stage, were reoriented to develop a system capable of removal of organic iodides instead of the original design to remove inorganic iodides. This change also required new investigations.

R&D required for the operation of Indian Point #2 has been completed. (See summary of application, section VII and the forthcoming answer to Board Question No. 16).