Chap 7

FN 37

July 8, 1966

MEMORANDUM

To : Dr. T. J. Thompson
Indian Point 2 Subcommittee Chairman

From : M. C. Gaske, ACRS Staff Signed Marvin C. Gaske

Subject : INDIAN POINT 2 SUBCOMMITTEE MEETING, JUNE 23, 1966

Attached is a draft of the Subcommittee minutes for the June 23, 1966 Indian Point 2 Subcommittee meeting which was held in Washington, D. C. Copies of the minutes are being distributed to the other ACRS members who attended the meeting, in the event they wish to comment, and the remainder of ACRS for information.

cc:

Remainder of ACRS
D. Duffey

Attached:

Minutes of MTG held 6/23/66

ACRS Job 429 Box 3 Shelf 11913
Correspondence 1/66 - 3/67
also
ACRS Job 1350 Box 17 Shelf 3982
Indian Point 2 11/65 - 12/69

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MINUTES OF
INDIAN POINT 2 SUBCOMMITTEE MEETING
Washington, D. C.
June 23, 1966

The purpose of the Indian Point 2 Subcommittee meeting held on June 23, 1966, in Washington, D. C. was to continue the Subcommittee review of Consolidated Edison Company's application for a construction permit for the Indian Point 2 facility. Present at this meeting were the following:

ACRS

T. J. Thompson
F. A. Gifford
S. H. Hanauer
H. W. Newson
N. J. Palladino
M. C. Gaske, ACRS Staff

Regulatory Staff

C. L. Allen
R. S. Boyd
L. I. Kopp
D. R. Mueller
P. E. Norian
F. Schauer
R. L. Waterfield
J. Proctor, Consultant

Consolidated Edison Company

W. J. Cahill, Jr.
H. W. Dierman
J. J. Grob, Jr.
J. A. Prestele
C. F. Soutar

LeBoeuf, Lamb & Leiby

E. B. Thomas
A. A. Upton

Consolidated Edison Consultants

G. Brown
C. R. McCullough

Westinghouse Electric Corp.

H. N. Andrews
E. S. Beckjord
A. R. Collier
H. J. Cordle
R. J. French
G. A. Harstead
W. Lester
J. S. Moore
R. C. Nichols
L. Porse
J. G. Russell
H. L. Russo
T. Stern

United Engineers & Constructors

S. B. Barnes
J. H. Hemsarth
R. O. Imhoff
D. Rhodes
D. Rose

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Executive Session

Dr. Thompson stated that the possibility of a reactor core melti ~ and forming a molten mass of material has previously been considered for fas. reactors. The problem of recriticality was the primary item of concern, and dividers have been placed !.low some fast reactors to seperate the molten fissionable material should a core melt down occur.

Dr. Hansuer called attention to the last paragraph in Third Supplement to: Preliminary Safety Analysis Report. This paragraph states that the cavity below the Indian Point 2 reactor pressure vessel will have the capability of preventing breaching of the containment by the molten core through the use of water in the cavity. He believed that, if Consolidated Edison could substantiate this claim, there may not be an unacceptable safety problem involved with core melt down at the Indian Point 2 facility. The question was raised whether Question 9 in the Third Supplement (Question 9 reads: Provide a discussion of the adequacy and reliability of the emergency core cooling system.) is present because of a desire on the part of the applicant or at the request of the Regulatory Staff.

It was reported that, on June 14, 1966, the DRL Staff contacted the ACRS Office regarding whether the Committee desired that written information be submitted by the applicants regarding Dresden 3 and Indian Point 2 relative to (1) the course of a core melt down accident without core cooling systems in operation and (2) the reliability of the emergency cooling systems. Following this, DRL contacted representatives of the applicant for each reactor and informed them that neither the ACRS nor DRL was requesting that written information be provided in regard to the above two items. Commonwealth Edison representatives indicated to DRL that no written information would be submitted regarding Dresden 3. Con Ed representatives stated that they might provide a limited amount of information in their Third Supplement which they were in the process of preparing for submission. DRL emphasized that neither they nor the ACRS was requesting such written information.

Prior to the June 23, 1966, Subcommittee meeting, Con Ed requested a list of items which might be discussed at the Subcommittee meeting. Such a list was provided to Con Ed, and a copy of this list is attached. Dr. Thompson stated that the Committee might want to write a letter to the Commission suggesting that safety research be performed regarding the problem of the consequences of a core melt down accident.

In regard to Item 3 of Attachment 1 (reactivity transients), Dr. Hansuer stated that the problem concerning reactivity transients had been that Con Ed's analysis was not sufficiently comprehensive and did not indicate how close the reactor might be to serious accidents if reactivity coefficients slightly different from those assumed were used in the analyses. Dr. Thompson provided the Subcommittee with a listing of characteristics of a number of boiling water reactors and pressurized water reactors (list forwarded to Committee members on 6/23/66). He pointed out that five Westinghouse reactors in succession have the same general reactivity characteristics and that San Onofre is the first in this series.

He indicated he believed that, if the ACRS approves the proposed operation of San Onofre, the Committee may have established a precedent for the allowable reactivity characteristics of subsequent Westinghouse reactors. He also pointed out that the Committee had recently acted in a favorable manner regarding the Brookwood application and that the Brookwood reactor has essentially the same reactivity characteristics as those of Indian Point 2.

Dr. Thompson stated he believed that the Indian Point 2 reactor could be made as safe relative to reactivity transients as any in operation. He said that Con Ed might have to retreat somewhat and that, perhaps, the ACRS should inform Con Ed that they must place a burnable poison in the core during the first part of the life of the initial core. He stated that Con Ed could place a boron stainless steel or other suitable poison in empty positions in the fuel assemblies and thereby make the moderator temperature coefficient significantly less positive during the initial operation of the reactor.

Regulatory Staff

DRL reported that a .5% reactivity insertion during full power operation would bring the hottest fuel pellets in the core to a just molten condition. A 1% insertion from full power would result in 11% of the core reaching a molten condition. If 11% of the core were to melt, it might jeopardize the integrity of the primary system.

The DRL Staff reported that their consultants for structural matters, Newmark and Hall, had most of their questions regarding the containment structure answered in Supplement No. 3. DRL understood that they still had one minor point which was unresolved, but DRL had not yet secured the final report from their consultants.

Mr. Boyd stated that the DRL Staff presently believes the proposed Indian Point 2 plant is an acceptable one. They consider Indian Point 2 to be a "suburban reactor". Dr. Hanauer inquired whether the Staff was satisfied it had enough information to convince itself that there is not a great extension of the containment through use of the isolation valve seal water system and that failure of the system will not cause greater leakage than the system is intended to eliminate. The DRL Staff indicated they were satisfied with the new seal water system which has been proposed by Westinghouse.

An intermediate size pipe break in the primary system is considered to be the worst type of pipe rupture. In event of a large pipe break, the system will rapidly blow down, and the low head safety injection system pumps can inject water into the reactor pressure vessel. If the pipe break is small, the high head safety injection pumps can handle the situation.

Mr. Boyd stated that Con Ed had elected to provide the information contained in Question 9 of the Third Supplement on their own initiative. The DRL Staff indicated that they did not believe there was a significant difference between Indian Point 2 and Dresden 3 regarding the consequences of a core melt down accident and the ultimate fate of the molten fuel. Mr. Boyd said he recognized the inconsistency of the Staff's present position that the core will melt for fission product release and metal-water reaction considerations but that the core will not melt and cause a problem as a result of the molten mass of material which would be present.

Mr. Proctor reported that he and another person at the Naval Ordnance Laboratory had both developed models regarding the effects of a pressure vessel rupture. Both of these models gave approximately the same results as the model used by Westinghouse. If a longitudinal rupture occurred, the pressure vessel would move outward approximately six inches and contact the concrete shielding located around it. Simultaneous rupture of the pressure vessel head bolts would result in the head passing through the top of the containment. Westinghouse still believes that such a rupture of the pressure vessel head bolts is incredible. If the head were to fly off in such a manner, the entire core would probably be lifted.

Mr. Norian of the DRL Staff reported that it appeared that the only pressure vessel failure that the reactor containment could not withstand is simultaneous failure of the pressure vessel head bolts.

Consolidated Edison Representatives

Mr. Beckjord stated that Westinghouse had concluded that the shielding structure around the reactor pressure vessel would not fail violently so as to cause missiles in the event of a longitudinal rupture of the pressure vessel. Westinghouse has assumed that a longitudinal crack would proceed down the wall of the pressure vessel but would stop at the bottom hemispherical region which is located outside the area which might be made brittle by irradiation.

Mr. Beckjord read a summary of Westinghouse's conclusions regarding rupture of the pressure vessel below the flange. The statement was as follows:

> "Control rod drive mechanism shield, reactor vessel shield, lifting rig, control rod drive mechanisms, studs, that part of vessel and flange above breaks, internals, core barrel and core are accelerated to a velocity of approximately 85 ft/sec at an elevation of approximately 25 ft above starting point. With no collision with crane, the above masses would rise an additional 112 ft approximately to a point about 21 ft below the top of containment. If a collision with crane occurs, the initial mass together with crane would rise between 30-39 ft based on inelastic impact, leaving a distance of 16-25 ft vertical rise to the point where the crane girder projection would touch the containment."

Westinghouse has studied reactivity insertion accidents involving the release of large quantities of sensible heat to the reactor coolant. These studies indicate that the pressure vessel might fail due to the quasi static pressure rise in the vessel. The quasi static pressure is the pressure which persists in the pressure vessel after the initial shock wave pressure rise.

If pressure were continuously raised in the pressure vessel, the concrete shielding surrounding the vessel would be contacted by the vessel and the shielding would eventually fail. The concrete shielding is capable of withstanding an internal pressure of approximately 750 to 800 psi. The design yield strength (650°F) of the pressure vessel head bolts is 9300 psi. Mr. Beckjord stated that the primary system piping and steam generator tubing would be expected to fail before the pressure vessel. He believes that hhe previous Westinghouse estimate of 500 - 550 ft rise of the pressure vessel head, in the event of simultaneous rupture of the pressure vessel head bolts, is conservative and that the vessel head might not rise that high. Mr. Beckjord said he would classify the possibility of pressure vessel failures as follows:

1. Simultaneous pressure vessel head bolt failure - incredible

2. Circumferential pressure vessel failure above the nozzles but below the flange - not a finite probability

3. Failure of the pressure vessel below the nozzles - extremely remote

Con Ed does not intend to place a microphone in the containment to assist in the detection of leaks within the containment. Failure of four adjacent head bolts would prevent reactor pressure from being maintained. Westinghouse has recommended to Con Ed that one half of the head bolts be inspected through use of ultrasonic testing and dye penetrant testing at each refueling. They also have recommended that all of the head bolts be visually inspected at each refueling.

The reactor core would be expected to rise if the pressure vessel head lifted from the top of the pressure vessel. Mr. Beckjord stated that the problem is alleviated somewhat because most of the water in the pressure vessel is located above the core. It was his opinion that the core would probably rise approximately 20 to 25 feet but would not leave the pressure vessel.

Dr. Gifford stated he believes the degree of concern relative to off-site emergency procedures should have some relation to the number of reactors present at a site. Con Ed has had some discussions with the New York State Police regarding arrangements for emergency off-site evacuation. Con Ed has not, however, tested emergency procedures for major accidents because of possible adverse public relations effects. No arrangements have been made with a nearby weather station for weather information to be provided in the event of an accident. Dr. Gifford suggested that such arrangements might be made and periodically tested.

Dr. Doan stated that Con Ed seems to have developed emergency plans forohandling releases of radioactivity at Part 20 limits rather than for the more significant releases which might result from possible accidents.

Dr. Hanauer pointed out that, if a group of intelligent people can not devise emergency plans for use during accident conditions, the reactor supervisor can not be expected to devise such plans on the spur of the moment at the time of an accident.

Dr. Hanauer inquired how well Con Ed would know what quantity of radioactivity was being released from the reactor stack in the event of a major accident. Con Ed representatives did not know the range of sensitivity of the presently installed Indian Point 1 stack monitor nor the range of sensitivity of monitors to be installed for Indian Point 2. Dr. Hanauer cautioned that the need for stack monitors with an adequate range for possible Indian Point 2 accident conditions was being indicated to Con Ed at an early stage.

The 47 psig containment design pressure is greater than the pressure in the water in the cooling coils of the air recirculation system located inside the containment. The pressure in the coils is approximately 20 to 25 psig. Dr. Thompson was concerned that, during accident conditions there might be a leakage of radioactivity to the river water circulating through the coils.

Dr. Hanauer pointed out that, in the event of a loss-of-coolant accident, three sets of pumps are required to function to remove heat through the residual heat exchangers. The service water system supplies coolant water from the river to the reactor plant component cooling heat exchangers. The component cooling heat exchangers in turn provide cooling water to the residual heat exchangers located inside the containment structure. These residual heat exchangers then remove heat from the containment internal recirculation system.

The isolation valve seal water system has been redesigned to incorporate the use of valves which seat at both the inlet and outlet portions of the valves. Water can be injected into the area between the two seats to prevent the leakage of gases through the valves when they are in a closed condition. These valves are being used, in lieu of the seal water system described in the application, in order to lessen the possibility that failure of the system might result in a leakage path from the containment. The applicant maintains that the new design is merely a change in detail and does not intend to submit an amendment describing the new system. Con Ed reported that there has been experience in the gas industry with the type of valves being proposed for use in the containment isolation valve seal water system.

Following a loss-of-coolant accident and injection of the contents of the refueling water storage tank into the containment, the water level in the containment structure will be approximately 3 feet above the floor level. The attached sketch was drawn by Westinghouse to show the level that the water would reach relative to the core.

Westinghouse supplied the following table of items which would be in operation if two of the three emergency diesels are operable:

INDIAN POINT PLANT SAFETY INJECTION OPERATION EMERGENCY POWER

(Injection Phase) Component	Horse Power	(Internal Recirculation) Component	Horse Power
4 fans	1400	2 fans	700
safety injection pump	350		
1 residual heat pump	250		
1 spray pump	350		
2 service water pumps	800	3 service water pumps	1200
		1 recirculation pump	350
		1 component cooling pump	200
Total	3150		2450

Approximately one hour after the occurrence of a loss-of-coolant accident, external pumping of water through the safety injection system and the containment spray system is ended, and a shift is made to the internal recirculation operational mode. Approximately 10 minutes are required to start and stop pumps, close valves, trip containment fans off the line, etc., during the shift. Dr. Hanauer inquired whether there would be adequate cooling through all of the charcoal filters if two of the air recirculation system fan coolers are turned off after the filters become laden with fission products. There is one charcoal filter in series with each fan. The discharge from each fan passes into a common discharge duct. Westinghouse maintained that, if two of the fans were turned off, there would be sufficient back flow through the other fans and their associated charcoal filters to prevent the charcoal filters from burning. The refueling water storage tank has a capacity of 320,000 gallons. It will take between one and two weeks to mix the boric acid solution required to fill the tank.

Con Ed will establish a criteria as to how much water must be in the refueling water storage tank to permit reactor operation to proceed. Westinghouse reported that, at a temperature just above freezing, the boron concentration in the refueling water storage tank would be approximately a factor of two below the saturation concentration. Dr. Newson inquired as to the reason the reactor pressure vessel is not located deeper in the cavity at the bottom of the containment in order to obtain better assurance that the core would be flooded if water is injected into the containment.

Mr. French of Westinghouse reported that burnable poison could be placed in the core without seriously affecting the power distribution. He stated that, if one weight percent boron stainless steel were inserted in the core, only 8% of the original boron in the stainless steel would be present at the end of the first core cycle. Westinghouse indicated that, if found necessary, a burnable poison could be added to the fuel assemblies near the time that reactor operation is to begin. Dr. Hanauer cautioned that, in view of the fact that rod programing is very important in the prevention of reactivity accidents, this matter will be looked at carefully at the operating license stage.

Westinghouse presented the following table:

LINES OF PROTECTION FOR CORE MELT THROUGH

1. Reactor coolant system integrity

   a. Designed to quality control standards
   b. Pressure vessel above NDT
   c. Ductile pipe

2. Safety injection system reliability

   a. Redundancy
   b. Five pumps (3 high head, 2 low head) not including recirculation pumps
   c. Eight injection points
   d. Three diesels, 3 instrument and control channels
   e. Reactor vessel and reactor coolant pipes supported for double ended pipe severance
   f. Failure analysis
   g. Recirculation

3. Safety Injection Performance

   a. Approx. 20% of installed deluge flow will prevent melt through
   b. 3 diesels - only two required

For the core melt through accident, Westinghouse has postulated that approximately 60% of the core is in a molten condition at the bottom of the reactor pressure vessel and that the bottom of the vessel is located in water. Several inches of insulation will be present around the pressure vessel, but this insulation is of a reflective type. It consists of aluminum foil placed between stainless steel sheets. Mr. Stern stated that, if steam is generated in the insulation, water can communicate freely through the insulation, and any steam formed would be readily removed. Assuming 60% of the core is molten and present in the bottom of the pressure vessel, and that the after heat generation rate is 2% of full power, Westinghouse believes that the core would not melt through the bottom of the pressure vessel. Westinghouse, however, had no experimental evidence to support their contention that there would be an adequate heat flow through the pressure vessel. It was pointed out that, if 60% of the core were actually molten, the fuel pellets remaining in a solid form would probably fall to the bottom of the vessel. The question was raised as to the effect the crust, which would form on the outer surface of the molten fuel, would have on heat transfer characteristics. Westinghouse has calculated that, if the core were molten and dropped into water, there would be a 15 psi rise in the pressure inside the containment. It was suggested to Westinghouse that they might wish to consider the melt through accident a general problem and publish a report on it that would not be related to a particular reactor.

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Westinghouse presented the following table which represents the sequence of events if all of the core is allowed to melt:

SEQUENCE OF EVENTS WITH NO ACTIVE QUENCHING OF CORE IN REACTOR VESSEL

blowdown	0 - 12 secs.
1st zone melts	145 secs.
bottom of pressure vessel dry	575 secs.
barrel melts	2300 secs.
melt through pressure vessel	2300 secs.

Executive Session

Dr. Thompson believed that Con Ed might be asked to consider:

1. Provision of means of insuring an adequate containment flooding depth to cover the core for any breaks in the primary system.

2. Placing a burnable poison in the core so that the quantity of boron in the coolant would be limited to the maximum of 1500 ppm during reactor operation.

3. Placing a large U tube between the Indian Point 1 and Indian Point 2 containments to provide for blowdown of pressure into the other containment in the event of excessive pressure in either containment.

4. Providing for cooling water to be added to the secondary side of the steam generators in order to condense steam and remove heat on the primary side in the event of a loss-of-coolant accident.

5. The addition of a holddown device to prevent the core from rising in the event of an accident.

Dr. Thompson decided to present the above in informal discussions with the applicant and to present these as his own individual ideas. The manner of handling each of the ten items in the list presented to Con Ed for oral discussion at the Subcommittee meeting was agreed upon.

Consolidated Edison Representatives

Dr. Thompson referenced the ten items in the list provided Con Ed for oral discussion at the Subcommittee meeting. He stated that Con Ed should be prepared to discuss these items at the full Committee meeting as follows:

1. Be prepared to discuss.

2. Be prepared to discuss.

3. Be prepared to discuss briefly and indicate the general means of retreat if the magnitude of the positive void coefficient is considered too large.

4. Be prepared to discuss.

5. Be prepared to discuss briefly.

6. Be prepared to discuss, especially the operating experience with the type of valves proposed for use.

7. Be prepared to discuss, particularly possibility that internal pressure of containment may exceed the pressure of the water in the cooling coils of the air recirculation system.

8. This item appeared to be reasonably well documented in the Third Supplement.

9. This item also appeared to be sufficiently well documented.

10. Emergency procedures would not be discussed at the full Committee meeting, but Con Ed should keep emergency procedures in mind for the operating license stage.

Dr. Hansuer stated that a question which should be explored is what would happen if the core melted through the pressure vessel. Westinghouse indicated they believed that, when the molten fuel melted through and hit the water located below, there would be rapid chilling and formation of solidified uranium oxide. Dr. Thompson emphasized the need for adequate information to be provided to the full Committee regarding the core melt through accident.

Westinghouse reported they are considering the installation of an open stainless steel tank at the bottom of the containment. The tank would be located below the pressure vessel and would be raised to permit circulation of water under it. The bottom of the tank would be lined with fire brick. In the event the core melted through the pressure vessel, approximately a one-foot thick mass of UO_2 would be deposited on top of the fire brick. Heat would be transferred by boiling water on top of the molten UO_2 and by the passage of some heat through the fire brick to water located below the tank. It was suggested that the bottom of the tank might have to be slanted to prevent the occurrence of steam blanketing below the tank.

At the end of the meeting, Con Ed revised their commitment to inspect one-half of the pressure vessel head bolts by ultrasonic and dye penetrant testing at each refueling. They agreed, instead, to inspect all of the bolts in this manner at each refueling.

Atts: (1) Subjects for Oral Discussion ...
(2) Chart

ACRS

* * *

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SUBJECTS FOR ORAL DISCUSSION
AT INDIAN POINT 2 SUBCOMMITTEE MEETING, JUNE 23, 1966

1. The possible course of a core meltdown accident without core cooling systems in operation and possible additional means of coping with this type of accident, if necessary. (Discuss, in particular, the ultimate fate of the molten fuel and any effect it may have on containment integrity.)

2. Reliability of the emergency core cooling system. (Discuss the differential pressure which may exist across core components during primary system blow down accidents and the ability of the components to withstand the forces involved. Discuss the effect that disarrangement of core components may have on cooling of the core by the emergency coolant system. Also, discuss the amount of pressure vessel movement which could occur without impairment of the proper functioning of the core emergency coolant system.)

3. Reactivity transients.

4. Current plans regarding designing against the longitudinal rupture of the reactor pressure vessel.

5. The capability for simultaneous operation of the emergency core cooling systems and the containment spray systems.

6. The isolation valve seal water system regarding its vulnerability to malfunctions which might violate containment integrity.

7. Reliability of the service water system during accident conditions.

8. The degree of inspection of the containment liner welds following fabrication.

9. Requirements, above those stated in Section III of the ASME Boiler and Pressure Vessel Code, which will be in effect for the Indian Point 2 reactor pressure vessel.

10. A brief description of all arrangements, procedures, and special equipment provided for use in emergencies which might affect persons located off-site. Discuss the degree to which emergency plans have been tested and the results of such tests.

ATT. 1 to Indian Point 2 Subcmte Minutes of Mtg held June 23, 1966

ACRS

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