PROD. & UTIL. FAC. 50-247

BEFORE THE UNITED STATES
ATOMIC ENERGY COMMISSION

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

Answers of Applicant to Questions Raised by Atomic Safety and Licensing Board on March 24, 1971

Part II

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KEY TO IDENTIFICATION OF QUESTIONS

(B) Question by Mr. Briggs

(G) Question by Dr. Geyer

(J) Question by Mr. Jensch

(Tr. 683) - Transcript Page 683

Question No. 3 (B) (TR. 683)

"Now, with regard to the reactor vessel and related somewhat to inspection, in reading the information that came from the construction permit stage, one, I think, is impressed that at that time, at least, there is some concern about the possibility of a rupture of the reactor vessel.

The Applicant provided special concrete structure, I will call it, shielding around the reactor vessel to prevent missiles. The Staff safety analysis indicated and the ACRS letter indicated that certain provisions were being made in the design of the plant to take care of meltdown fuel and this could be expected in the event of a rupture in the reactor vessel.

In ACRS documents prior to that time a concern was expressed. Yet, as the plant comes up for an operating license, it is indicated that there is now no concern about the rupture of a reactor vessel, that this is not a design basis accident that needs to be considered and, in fact, there is no provision for handling the meltdown of the core should such an accident happen.

I think it is important to justify this change in outlook that has occurred between 1965 and 1966 when a construction permit was issued and the present stand."

Answer:

No provisions were ever made or considered for Indian Point Unit No. 2 or for any other water reactor of which the Applicant is aware that would take care of fuel meltdown following the rupture of the reactor vessel. It is the position of the Applicant that the conservatism in the design and care taken in the manufacturing process, strict quality control and quality assurance during every facet of the design and manufacturing process combined with careful operation and a responsible in-service inspection program, both conforming to Technical Specification requirements, eliminates the probability of reactor vessel rupture and, hence, its consideration as a design basis event.

The crucible was proposed as a back-up to the Emergency Core Cooling System at a time when that system was not designed to prevent fuel clad melting for large breaks in the reactor coolant piping. The system as then designed permitted higher peak fuel clad temperature than can occur with the present improved design. In addition, the present Emergency Core Cooling System design includes valving and piping modifications which give capability to maintain core cooling and containment cooling in the event of a possible failure in the safety injection system or service water system for the long-term after a loss-of-coolant. The purpose of the crucible was specifically to handle molten fuel should such occur and melt through the reactor vessel after a large loss-of-coolant accident.

After the Indian Point Unit No. 2 construction permit was granted the design of the Emergency Core Cooling System was changed to increase its capability to reduce the maximum fuel clad temperature and restrict the amount of metal-water reaction which can occur in a loss-of-coolant accident and thereby maintain the effectiveness of the Emergency Core Cooling System. The concurrent development of the accumulator system by Westinghouse made this technically feasible.

At the same time, the development of a crucible to handle the potential effects of molten fuel in the containment was found to be impractical and because the improvements in the Emergency Core Cooling System removed concerns about fuel clad melting which led to the adoption of the crucible concept

in the first place, it was therefore proposed and the AEC after its review concurred in not providing the crucible.

"In connection with the emergency core cooling system, as I read the Staff safety analysis and the ACRS letter for 1966, the emergency core cooling system as proposed at that time was inadequate. The flow from the system was going to have to be increased and increased to the extent that a meltdown of the core could not occur.

This at least is my reading of the reports at that time.

In addition to that the Applicant was going to provide and the papers seemed to indicate that it would be nice to provide for containment, at least, in the event this emergency core cooling system or one that was supposed to fail actually did fail and this was the reason for putting the crucible below the reactor vessel.

This was no longer considered necessary. In other words, no back-up for the emergency core cooling system seems to be considered necessary. Though there may be very good reasons for this, I think it would be desirable to discuss at the hearing more about what work was done on the design of that core catcher, I will call it, because this is stated in reply to the extensive design work that was done.

Give additional information concerning the reasons for removing this device, even though it was provided only as a back-up to a system that was not supposed to fail."

Answer:

The reasons for elimination of the crucible are discussed in response to ASLB Question 3 (B) (TR. 683). Design work done on the crucible prior to its elimination was in the areas of layout, structure, heat transfer and materials selection. The result of this work is described in Section 7 of Supplement 7 to the PSAR of Indian Point Unit No. 2 (attached).

7. REACTOR PIT CRUCIBLE

7.1 SUMMARY DESCRIPTION

The reactor pit crucible was proposed in Supplement 5 to serve as back-up to the emergency core cooling system in the event that the core might melt and deposit in the reactor vessel cavity. The reactor pit crucible, Figure 7-1, is a refractory lined vessel with a sloped bottom supported and elevated from the cavity floor by structural members which allow free flow of water beneath the vessel and steam separation by way of the space between the vessel sides and the concrete reactor cavity walls. The crucible is located directly below the reactor vessel and in-core instrumentation guide thimbles, and extends into the access tunnel as shown on Figure 7-2. The capacity of the crucible is sufficient to contain all of the fuel (UO_2), fuel assembly grids (Inconel), fuel assembly end fixtures (stainless steel) and portions of the lower core support structure and reactor vessel bottom head.

The addition of the accumulators to the redundant and independently protected emergency core cooling systems relegates the reactor pit crucible back up to a role which will never be needed, even under the worst accident hypothesis. Core thermal transient studies (Section 1, Supplement Six) analytically demonstrate the capability of the emergency core cooling system, with its increased capacity, in preventing any clad melting for a loss of coolant accident even when considering complete rupture of a reactor coolant pipe.

The engineering approach to design of the reactor pit crucible was based on the following generalized assumptions.

1. The reactor pit is submerged with borated water from the break in the reactor coolant system and introduction of containment spray water from the refueling water storage tank (350,000 gals capacity).

2. Molten agglomerate including UO_2 solidifies on contact with the water cooled refractory.

With solid external boundarys of the agglomerate mass, an insulating barrier is obtained which essentially lowers the interface temperatures in the refractory to plate region.

Thermal analysis presented in Supplement 5 indicated refractories with service temperatures in the range of 3000°F are adequate. Of the many materials available today, silicon carbide brick and a high alumina brick are presently being considered for use in Indian Point No. 2. Chemical and Physical characteristics of the above materials are shown in Table 7-1.

Both materials are available in the same fundamental geometeric shapes and are suitable for attachment to steel back up plates. It is presently planned to cover the refractory with a ten or twelve gauge stainless steel liner to preclude material damage and moisture absorption during plant life time.

TABLE 7-1

CHEMICAL AND PHYSICAL CHARACTERISTICS
OF REFRACTIES

CHEMICAL COMPOSITION	Percent Alumina Brick	Silica Carbide
Al_2O_3	89.0	0.78
SiO_2	9.0	11.75
SiO	-	86.10
Fa_2O_3	0.25	1.05
TiO_2	0.10	-
CaO	0.02	0.21
MgO	0.02	0.10
Na_2O	-	Trace
K_2O	-	None
B_2O_3	-	-
PHYSICAL CHARACTERISTICS		
Bulk Density		
gms/cc	3.01-3.06	2.57-2.65
lbs/ft^3	188-191	161-168
Porosity-Percent	14-16	14-17
Modulus of Rupture-psi	2800-3600	3000-4000
Cold Crushing Strength-psi	14-18000	14-18000
Use Temp.-°F	3100	3100
Coef. of Thermal Expansion (per °F - 212 to 1800°F)	4.2×10^{-6}	2.6×10^{-6}
Thermal Conductivity ($BTU/hr/ft^2/in/°F$)	$\cong$ 16	105

7.2 DESIGN BASIS

The crucible is designed to contain the residue from core meltdown and vessel melt through, thereby preventing contact of the meltdown residue with the containment.

The load criteria for the design of the refractory lined crucible are based on a core and reactor vessel meltdown residue of 512,000 lbs. determined as follows:

		lbs.
1.	Reactor fuel (UO_2)	220,217
2.	Zr (as ZrO_2)	
	Fuel assembly girds (Inconel)	
	Fuel assembly end fixtures (Stainless steel)	61,100
3.	Lower core support structure (Stainless steel)	166,000
4.	Reactor vessel bottom head (Carbon steel)	64,500
	Total	511,817

The limiting case for crucible structural design is the assumption of meltdown residue collecting in a cone shaped mound with a 1:1 height to diameter ratio and a density of 750 lbs/ft^3 which results in a core volume of 683 ft^3. The weight of the meltdown cone over its area results in a load of approximately 8300 psf on the crucible under the reactor, a value well within the strength of the structure at its use temperature.

The dynamic loading of the crucible structure associated with vessel melt through and Class I seismic criteria, as defined in Supplement Two to the Preliminary Safety Analysis Report, have been applied to the structural design.

The cooling mechanism of the crucible was analyzed in Section 2 of Supplement No. 5 to the Indian Point Unit No. 2 Preliminary Safety Analysis Report. The principle consideration for crucible thermal integrity is that the steel vessel be maintained at temperatures below which its strength satisfies the load requirements previously listed. The major item affecting temperature in the steel vessel is the heat flux which determines surface temperature and temperature rise through the metal. The heat flux is dependent on the "melt" volumetric heat generation rate, and conductivities of the solid and liquid UO_2 phases which with the UO_2 vaporizing temperature and the refractory melting temperature establish the thickness of the solid and liquid UO_2 layers which conduct heat into the refractory.

In Section 7.4, conservative estimates for the above parameters have been used in a steady state thermal analysis which establishes that the mode of heat transfer at the vessel water interface is nucleate boiling and the temperature in the carbon steel crucible vessel would be maintained below 800°F. In Supplement 5 the transient effect of initially molten UO_2 contacting the crucible was determined to result in a monotonic heat up of the system verifying the applicability of the steady state thermal analysis.

7.3 MECHANICAL AND STRUCTURAL DESIGN

7.3.1 CRUCIBLE (See Figures 7-2 and 7-3)

The bottom of the crucible is composed of 12 inch wide flange sections sitting on concrete pads to form a trough with a minimum angle of 20° from the horizontal up to a maximum of 37° at toe of boot. One inch carbon steel plates are welded to the wide flange sections; this plate supports the refractory.

The side walls of the crucible under the incore instrumentation leads consist of 18 inch wide flange columns resting on concrete and bearing against concrete side walls. Where crucible side walls taper down to 8 inch wide flange sections due to clearance required for incore instrumentatic the columns are 1 inch plate box sections. One inch steel plates are welded to these vertical columns. The side walls extend approximately 11 feet (up to El. 29'-0", top of crucible) above the lowest point of the concrete.

The side walls under the reactor, consist of 2 inch thick plate against the concrete, 18 inch wide flange sections welded to the plate and then a 2 inch layer of plate welded to the 18 inch wide flange columns. These walls extend approximately 12 1/2 ft above the lowest point of concrete. The side walls will have 2 inch T-bars welded vertically at 2 ft. center to center spacing to support the 4-1/2 inch refractory.

The bottom plate is designed for approximately 8300 psf under the reactor and based on approximately 685 cubic feet of molten material. A loading of approximately 4300 psf is assumed at the toe of the boot. The temperature of the plate as a result of meltdown is considered in determining allowable plate stresses.

There are four - 8 inch diameter pipes encased in concrete around the reactor section that will carry water from elevation 46' - 0" to the underside of the crucible by gravity flow.

The side walls of the crucible will have a baffle located around the top extending out from concrete wall to direct falling material inward toward the crucible and away from the cavity annulus.

All structural steel sections and plates will be in accordance with ASTM A36 specification.

All welding will be in accordance with the latest edition of American Welding Society D1.0-66.

Section 2 of the Fifth Supplement to the PSAR describes the mechanisms of heat transfer which govern the design of the reactor pit crucible. A hypothetical accident is postulated in which total failure of core cooling systems results in melting and migration of the core material, lower core support and reactor vessel bottom head, to the crucible. In this configuration, the core mass is surrounded by water resulting from reactor coolant loss and containment spray. Self heating of the core mass is only partially dissipated by conduction through the solidified outer crust. The remainder is transferred by expulsion of UO_2 vapor and liquid through breaks in the crust created by UO_2 vapor pressure. Quenching of these eruptions by the surrounding water causes steaming in the reactor cavity which is relieved through the access openings to the reactor loop compartment where it is ultimately condensed by containment spray or the air recirculation coolers.

It is the specific purpose of the crucible in this hypothetical situation to maintain insulation of the foundation mat and liner from the core mass. A water space is provided surrounding the crucible shell for the purpose of cooling the steel plate and refractory insulation of the crucible. Boiling at the plate surface maintains a temperature gradient through the crucible structure which is definable in terms of the volume heat source, melting points and thermal conductivities of the materials. A satisfactory evaluation is obtained when the average temperature of the steel plate is within the range where adequate structural strength is assured and the heat flux at the water/steel interface is within the nucleate boiling range.

Heat Transfer at the Crucible Wall

The transient and steady-state heat transfer models with which the temperature distribution in the crucible wall was determined are described in Supplement Five, pages 2-4 to 2-10. Further consideration of the problem during subsequent engineering studies has confirmed the applicability of these models, subject to confirmation of the thermal properties of the materials used. In the present supplement, the results of a recalculation of temperatures with parameters reflecting design are presented.

The following refinements of input parameters are here incorporated:

1. Internal heat generation rate (q''') of the UO_2 has been selected as 0.4×10^6 Btu/ft^3 hr based on the residual heat rate (1.9% of the full power value) at an elapsed time of 2000 seconds after shut down. Noble gases, halogens, and 5% of the cesium fission products are assumed stripped from the fuel. The remaining decay heat sources are assumed to be present in concentrations representative of the leading 10% of the core, having an average power density which is 1.6 times the core average.

2. A value of 5.2 Btu/ft-hr-F° was assumed for the effective conductivity of the molten UO_2 below the vaporization temperature. This was the "worst case" assumption used in the Fifth Supplement analysis.

3. Solid UO_2 conductivity was taken as 2.5 Btu/ft-hr-F°, again the worst case value, to maximize the heat flux to the crucible wall.

4. A maximum UO_2 liquid temperature of 8000°F was assumed, corresponding to the vaporization temperature of UO_2 at 4000 psia.

5. Refractory conductivity values of 1.33 and 8.75 Btu/ft-hr-F° were assumed, representing alternate material selections of Cerox 700 (alumina) and Kellog 3 AD (silicon carbide), respectively. In both cases the thermal analysis was based on the assumption that above 3000°F the refractory was removed from the slab, reducing the thickness of the refractory layer until only material below 3000°F remained. In this manner a conservative allowance was made for possible eutectic formation which might reduce the protection of the steel afforded by the firebrick.

The maximum heat flux to water estimated in these calculation is 1.28×10^5 Btu/ft^2 hr. No higher value can be expected, because:

a) Higher conductivity of refractory is not obtainable with the materials under consideration.

b) Maximum values of UO_2 solid and liquid conductivities are assumed.

c) Maximum value of UO_2 liquid temperature is assumed.

d) Lower values of the parameters listed in (a-c) give rise to less UO_2 contained in the conducting layer, hence a lower heat flux.

The maximum heat flux of 1.28×10^5 Btu/hr ft^2 is within the range where pool boiling heat transfer will maintain the water-side wall surface temperature below 330 °F with a conservative heat transfer coefficient of 4000 Btu/ft^2-hr-°F. DNB heat flux at this condition would be expected to be approximately 200,000 Btu/ft^2 hr. The calculation has thus established a stable heat transfer mode capable of dissipating the largest heat flux which can be delivered to the crucible/fuel interface. Table 7-2 summarizes the region thicknesses and interface temperatures for both the insulating materials under consideration.

At a heat flux of 1.28×10^5 Btu/hr ft^2, the calculated steam separation rate at the periphery of the crucible is a maximum of 0.122 lb/sec. per foot of perimeter. A minimum of 18-in radial clearance is provided at all points, sufficient to pass this flow of steam with a superficial velocity of about 1 ft/sec, hence there would be no tendency to vapor-bind the region under the crucible.

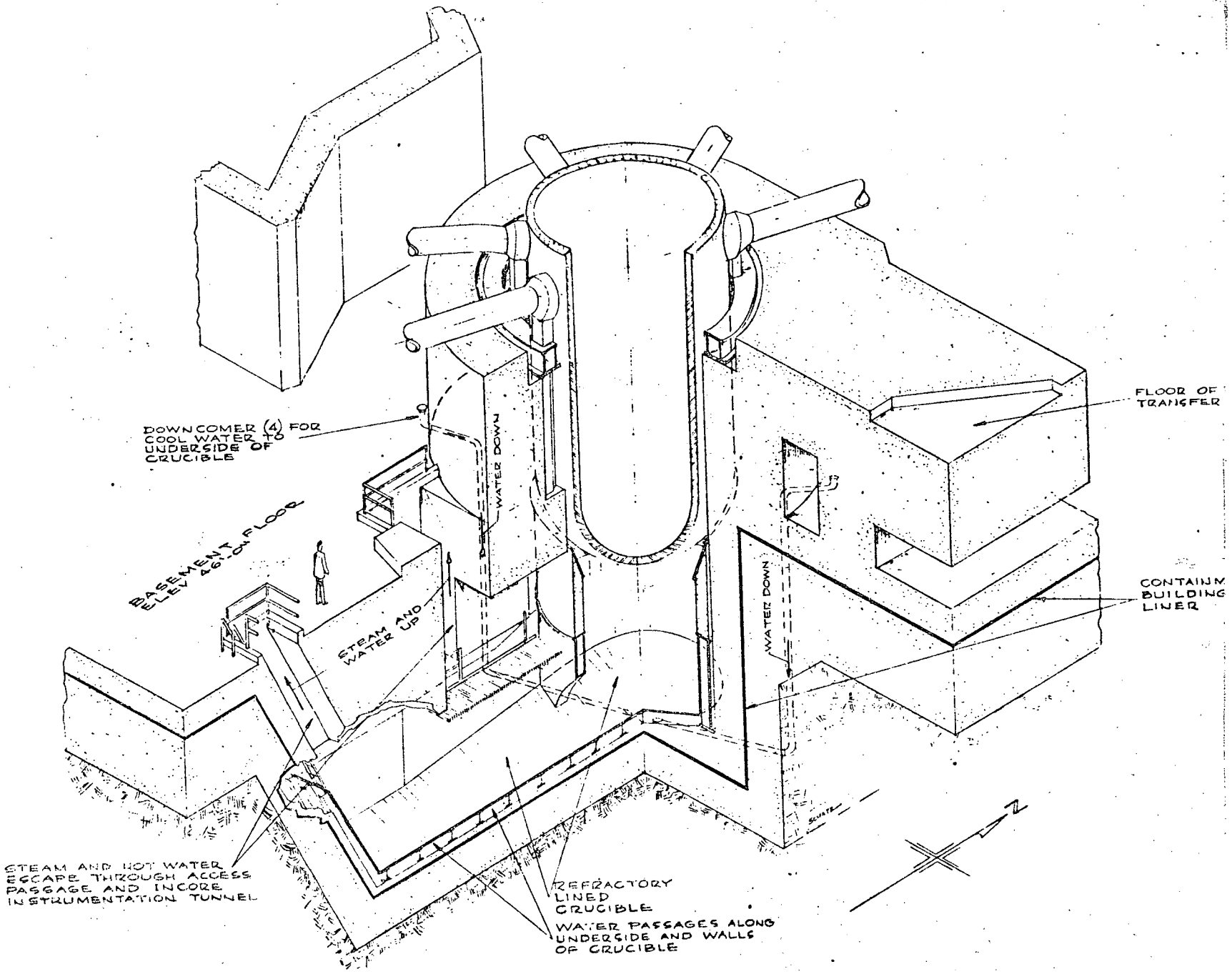
The total rate of steam formation from the entire core mass, if it were contained in the crucible cavity and assuming saturated water were returned from the containment floor to the cavity, would be about 65 lb/sec.

The available area to pass this steam flow is approximately 30 sq. ft., comprising the openings for personnel access and in-core detector conduit into the reactor cavity. The superficial velocity of steam rising through these openings is about 32 ft/sec. The calculated pressure drop for this steam flow is less than one inch of water, hence the back pressure created is of no consequence in limiting the natural convection flow of water through the downcomers in the reactor cavity.

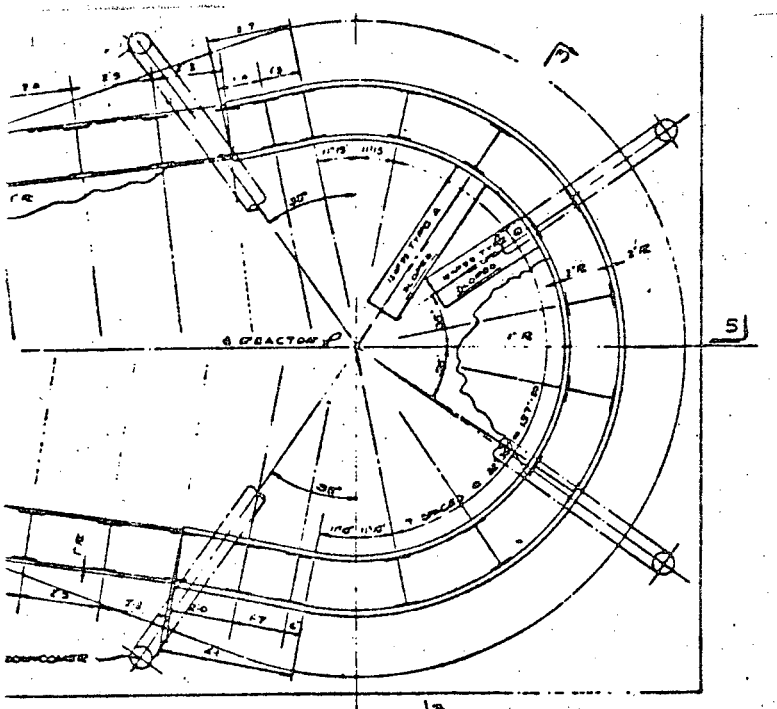
TABLE 7-2

SUMMARY OF THERMAL ANALYSIS

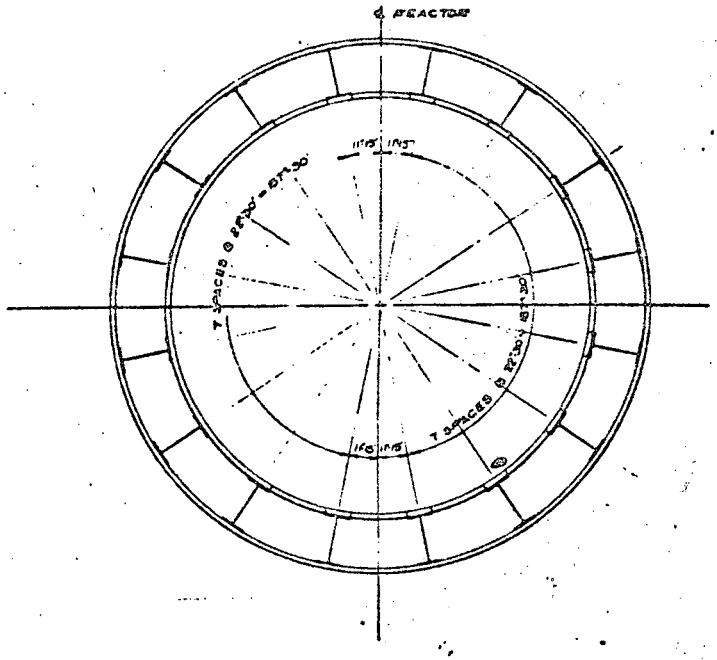
Volumetric Heat Generation Rate (Btu/ft^3-hr)		400,000
UO_2 Vapor Temperature (°F)		8000
Liquid UO_2 Conductivity (Btu/ft-hr-°F)		5.2
Liquid UO_2 Thickness below Vapor Temperature (in.)		3.4
Solid UO_2 Conductivity (Btu/hr-ft-°F)		2.5
Solid UO_2 Thickness (in.)		0.5
Heat Flux Through Crucible (Btu/ft^2-hr-°F)		128,000
Refractory Material	Cerox-700	Kellog 3 AD
Refractory Conductivity (Btu/hr-ft-°F)	1.3	8.8
Refractory Thickness Below 3000°F (In.)	1.88	0.29
Maximum Steel Plate Temperature (°F)	800	800



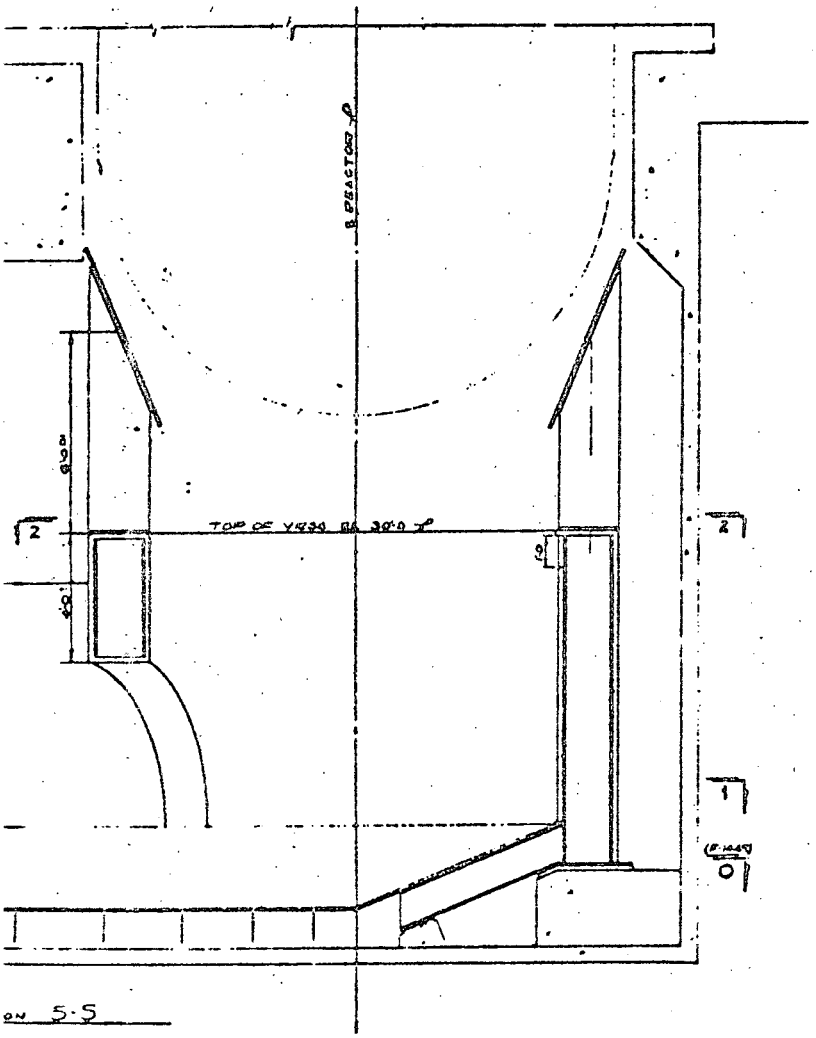
REACTOR PIT CRUCIBLE
FIG. 7-1



SECTION 1-1
SCALE 1/8" = 1'-0



SECTION 2-2



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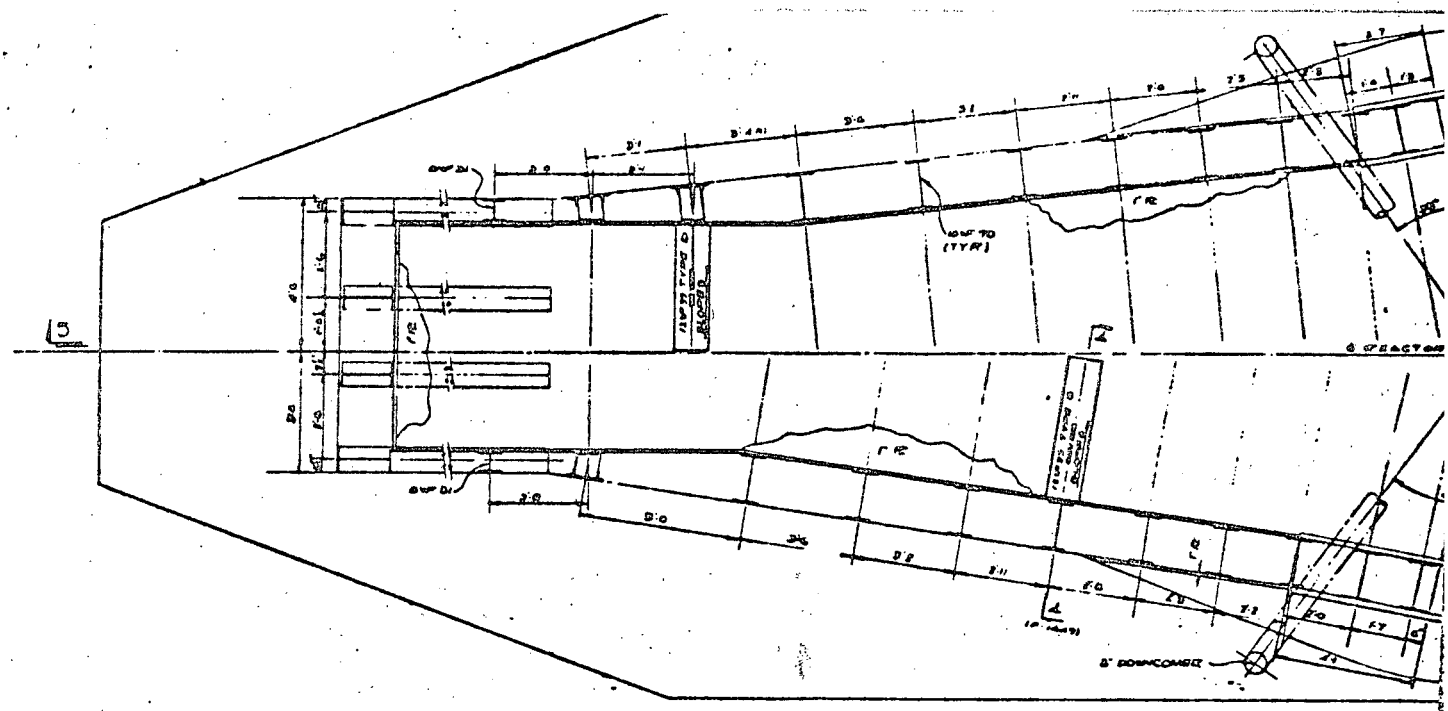
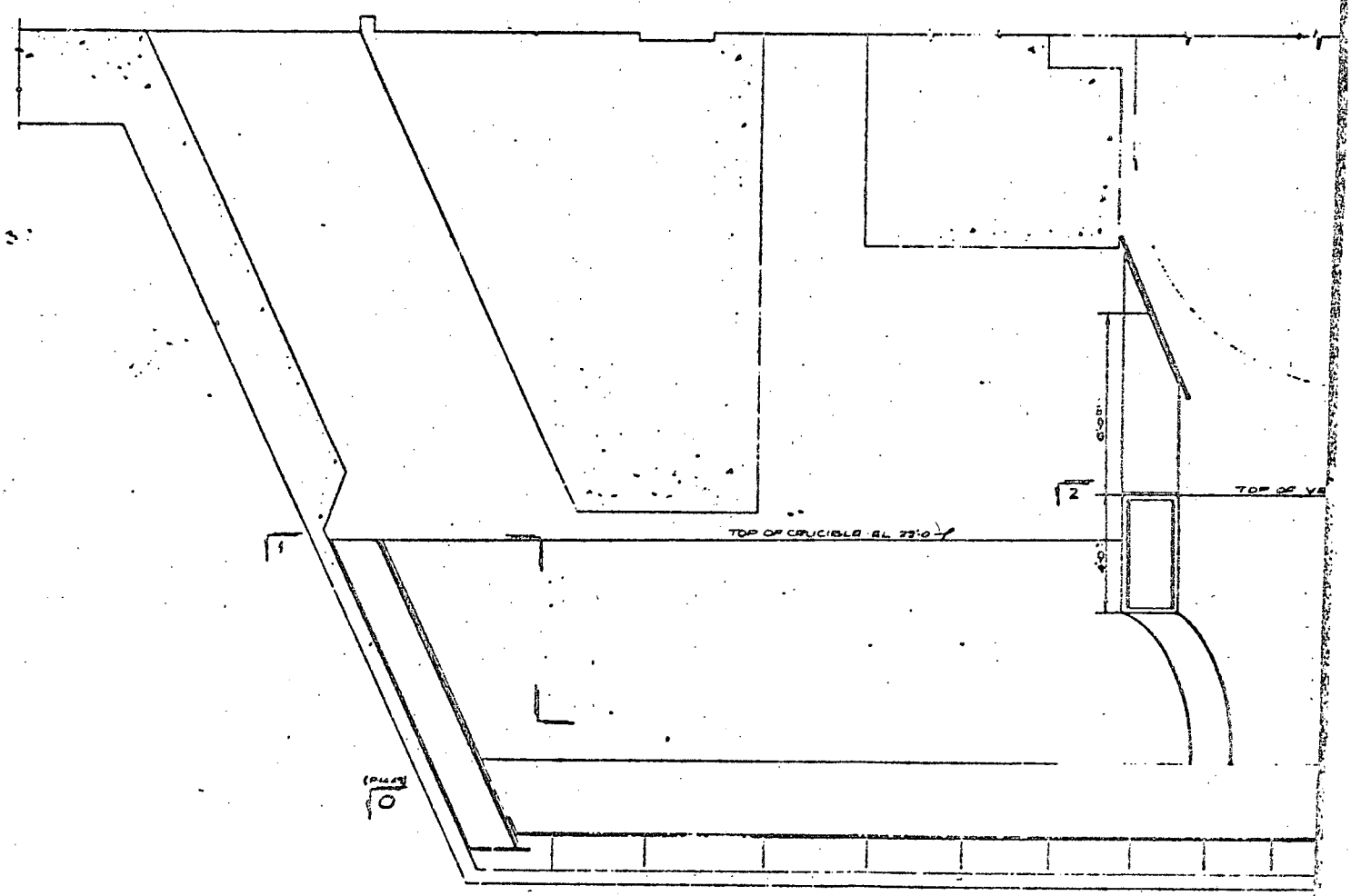
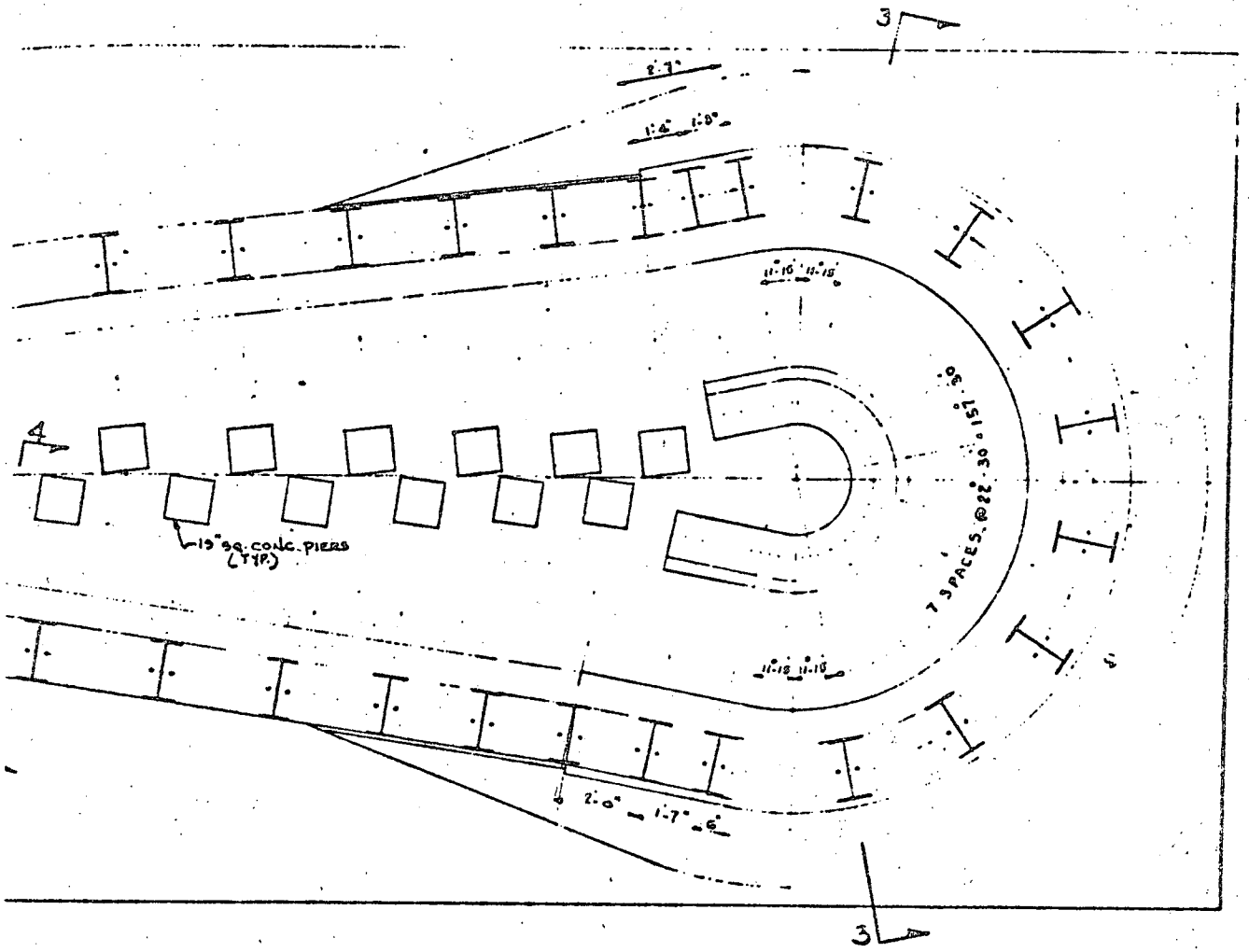
REACTOR PIT CRUCIBLE STEEL DETAILS
FIG. 7-2



Section 1-1
SCALE 1/8" = 1'-0

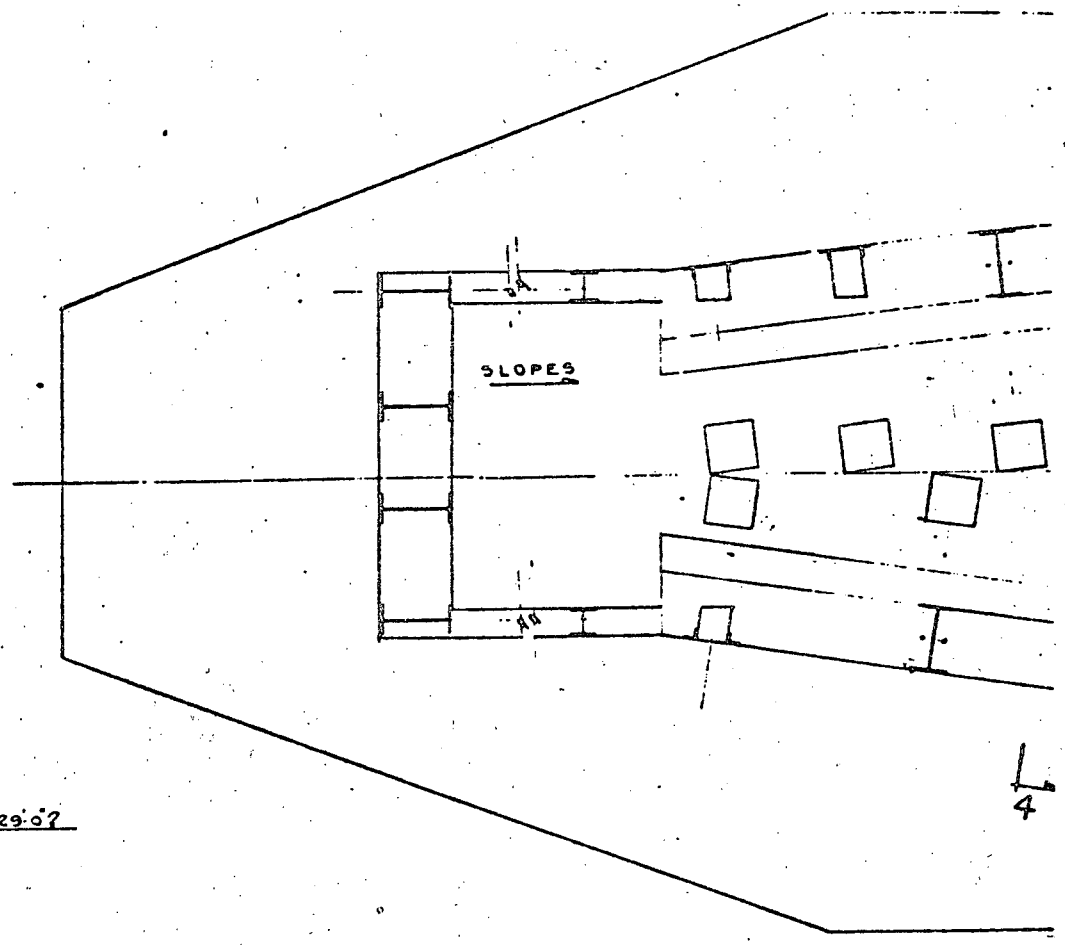


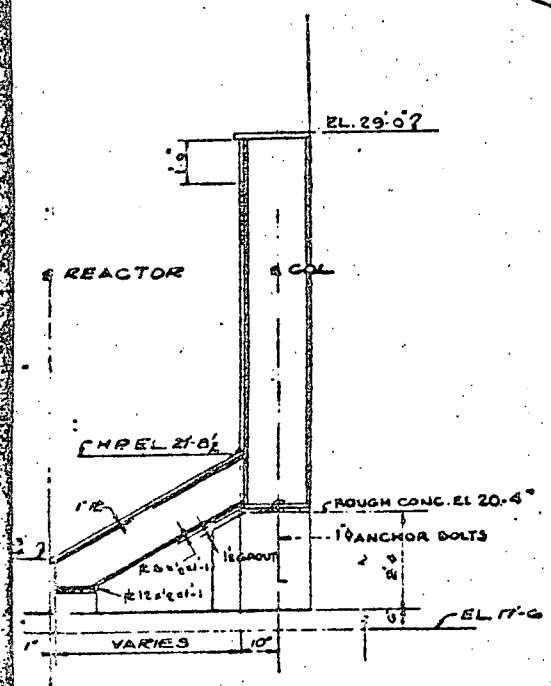
Section 5-5



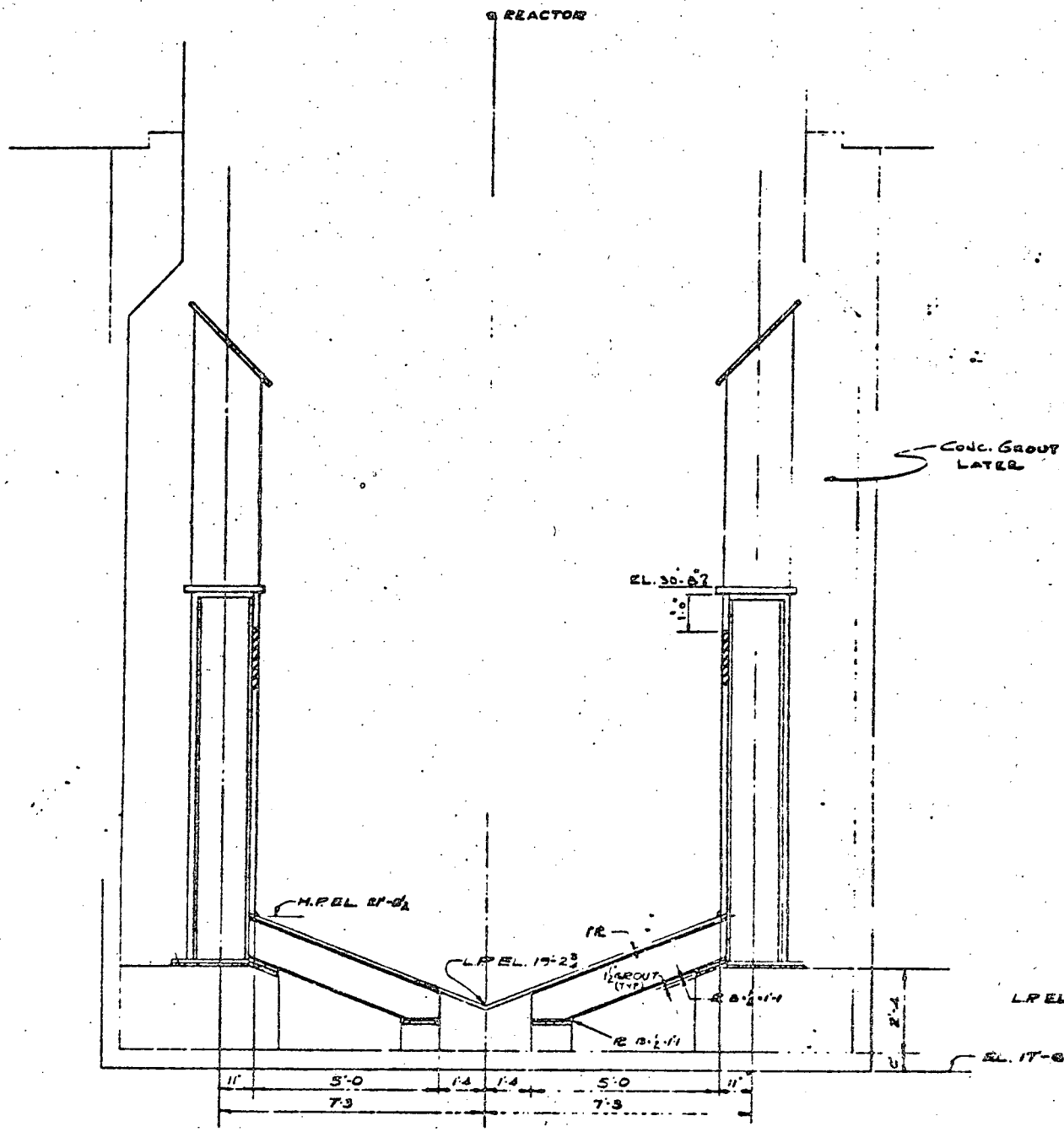
PLAN O-O (F-1447)

REACTOR PIT CRUCIBLE SECTIONS AND CONCRETE OUTLINE
FIG. 7-3


SLOPES
4


EL. 29'-0"?
℄ REACTOR
℄ COL
℄ HP EL. 21'-8½
ROUGH CONC. EL 20'-4"
1" ANCHOR BOLTS
1½ GROUT
EL. 17'-6
VARIES
10"

SECTION 4-4


REACTOR
CONC. GROUT LATER
EL. 30'-8½"
H.P. EL. 21'-8½"
L.P. EL. 19'-2¾"
1½" GROUT (TYP)
L.P. EL. 19'-2"
EL. 17'-6"
11"
5'-0
1'-4
1'-4
5'-0
11"
7'-3
7'-3

SECTION 3-3

Question No. 5 (B) (TR. 685) ASLB 3/24

"In connection with the emergency plans, there are procedures that are to be followed in the event of an emergency. These are procedures that have been provided by the Applicant and others provided by the State of New York. If the Applicant has anlyzed an accident, one that would involve extensive threat of radioactivity such as the State of New York to be called in, we would like there to be some discussion of the accident and the time that is involved.

Certainly the amounts of time required to notify people and take measurements. I have seen no description of a typical accident; I should call it an accident that is not typical, one that involves a considerable threat of radioactivity, and the time allowed for carrying out these operations according to the Staff's safety analysis; within two hours at the site boundary one could approach the 10CFR100 limits under certain conditions and 12 hours seems to be a fairly short time to carry out all the emergency actions called for in the emergency plan.

We would like to have some discussion about the kind of accidents that have been analyzed and the amount of time considered to be available for carrying out these plans and how they compare with this two-hour business at the site boundary."

Answer:

The Applicant has designed its Radiation Contingency Plan to be responsive to a variety of accident situations, which are generally described in the three categories referred to in FSAR Question 12.5, Section 3.0 beginning on Page 5.

The Applicant has discussed with the State of New York and the AEC Regulatory Staff the spectrum of accidents with which the emergency plan for the Indian Point site is designed to cope. It is the Applicant's judgment that all of these accidents either would not result in releases of radioactivity off-site or would produce off-site doses significantly less than those contemplated by the AEC's reactor site criteria (10CFR100).

Taking the case of the hypothetical occurrence of the most serious accident with which the emergency plan is designed to cope, i.e., a double-ended rupture of the largest primary coolant pipe and consequent release of radioactivity into the containment, the Applicant would be required to carry out the following activities related to the emergency plan:

a. Notify the Department of Health through the officer on duty at the 24-hour emergency number (518-457-2200).

b. As indicated in Applicant's answer to Question No. 14 of the Board (offered in evidence on March 24, 1971), the Applicant will also provide the Department of Health with 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 along with the seal water injection system is designed to prevent

such containment leak rate, because the field survey monitoring data, 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 is is considerably below the 1/10 of a percent per day assumed, the calculated doses will be adjusted accordingly.

It is estimated that the foregoing information could be provided to the Department of Health within one-half hour after the onset of the accident.

Other emergency notifications and activities of the Applicant in connection with a design basis accident are set forth in FSAR Question 12.5, on Pages 23 - 30.

The Applicant would institute off-site radiation surveys following the above-described accident. Initial radiation surveys off-site would be made by Con Edison's plant health physics survey team who would use a survey truck with two-way radiocommunication to the control room. These personnel would monitor airborne radioactivity and direct radiation downwind of the site. It is anticipated that initial results of these surveys would be available in approximately two hours. Other emergency monitoring assistance (e.g., from the AEC) should be available within four hours. The times involved in carrying out radiation surveys by off-site support groups would vary, of course, with the extent and nature of the radioactivity involved.

"There is a question concerning whether the releases are as low as practicable or whether they are a small percentage of the MPC value. I think at the hearing we will go into some extent to the provisions that have been made for controlling the routine releases from the plant and into the question as to whether they are indeed as low as one should expect, what kinds of modifications might be required to reduce them further and whether there would be any real advantage to such reduction.

In other words, whether the reduction would be so significant as to be concerned. I don't believe I have any other points to consider."

Answer:

The basic waste processing system design in the Indian Point Unit No. 2 plant to collect radioactivity has been discussed in Chapter 11 of the FSAR. In the plant, various radioactive fluids are collected and processed by either demineralization, filtration, evaporation, or some combination of all of these.

Based on the estimates given in the FSAR of the quantity of liquids to be handled in the WDS, and the design objective for equipment performance, Table 1 has been prepared to indicate the design objective of radioactive substances to be released from the Indian Point Unit No. 2 plant. The estimated activity to be released from the plant through the liquid system, exclusive of tritium, over the year has been estimated at .025 curies per year. This represents on an average annual basis a concentration when mixed with minimum circulating water flow of less than 1/50,000 of 10CFR20 regulations for all isotopes, exclusive of tritium. Tritium release as calculated on the basis of knowledge available at the writing of the

FSAR was estimated at 4240 curies per year. This total quantity of tritium on an average annual releases basis is less than 1/1000 of 10CFR20 MPC values. Operating experience with zircaloy cladding has indicated that this tritium figure will be substantially reduced. In any case, some tritiated liquid will be discharged from the plant in the form of HTO or T_2O, as there is no practical means of separating tritium. Concentrations quoted above are in the circulating water discharged from the plant, and further dilution will occur upon mixing with the Hudson River which flows past the plant.

Table 2 has been prepared to indicate the estimated gaseous releases from the Indian Point Plant. It will be noted that approximately 10,000 curies per year of gases will be released from the plant throughout the year. The estimated integrated whole body dose for a person residing continually at the site boundary would result in less than 20 mr/yr. This level of exposure represents less than 1/5 of the estimated annual dose from natural background source. Based on the above, it is clear that with this level of releases the plant will be operating at a level which is a very small percentage of the existing 10CFR20 regulations.

In order to ensure that the design objectives are realized, several modifications to the Indian Point Unit No. 2 plant are being made to the radioactive waste processing systems. Several of these modifications represent mechanical modifications to basic plant equipment where leakage of fluid has been a problem.

Specifically, modifications will be made to reciprocating charging pumps to collect leakage from these pumps and return it to the CVCS and secondly, modified bellows seal assembly is being provided on the pressurizer spray valves to eliminate the leakage being experienced from these components. Both of these items have been problems on existing plants and only as a result of recent testing and development has it been possible to incorporate these features in the Indian Point Unit No. 2 plant. These two specific modifications will reduce the amount of primary coolant leakage, the processing load on the evaporators, and consequently, the amount of activity being released.

In addition, modifications will be made to the waste disposal evaporator and are presently in design. These modifications are being made to improve the capacity and to achieve the decontamination factor as originally indicated in the equipment specification. These modifications are expected also to increase substantially the operability and consistent performance of the unit.

As a back-up measure to reduce the release from the plant in the event the evaporator by itself does not produce a sufficiently high decontamination factor, a polishing demineralizer/filter installation is being designed and will be procured for waste evaporator condensate cleanup. The function of these units will be to provide additional polishing to the distillate from the waste evaporator prior to discharge. The polishing demineralizers will provide an additional reduction in activity

released from the plant and will be installed in the event the modifications to the evaporator do not provide sufficient reduction in activity release.

A charcoal filtration system will be installed in the plant vent to significantly reduce any gaseous releases of iodine, which might occur from containment purge in the event of the existence of radioactive iodine in the primary coolant concurrent with primary leakage to containment atmosphere.

An intertie will be provided between the Indian Point Unit No. 2 steam generator blowdown lines and the new Indian Point Unit No. 1 blowdown purification system. This intertie will serve to significantly reduce liquid releases in the event of steam generator leakage.

It is anticipated that the modifications referred to in the last two paragraphs will be completed by the end of the first refueling outage.

TABLE I

Estimated Liquid Radioactivity Release From
IPP #2 and Resultant Concentration In Plant Effluent

Isotope	Annual Release $\mu C/yr$	Concentration In Circ. Water Disch. Canal $\mu C/cc$	Max. Per. Concentration (10CFR20) $\mu C/cc$	Fraction of MPC
Mn-54	9.08×10^{-1}	6.05×10^{-16}	1×10^{-4}	6×10^{-12}
Mn-56	1.94×10^{1}	1.30×10^{-14}	1×10^{-4}	1.3×10^{-10}
Co-58	2.34×10^{1}	1.56×10^{-14}	9×10^{-5}	1.7×10^{-10}
Co-60	2.99	1.99×10^{-15}	3×10^{-5}	6.6×10^{-9}
Sr-89	9.09	6.06×10^{-15}	3×10^{-6}	2×10^{-9}
Sr-90	4.46×10^{-1}	2.97×10^{-16}	3×10^{-7}	1×10^{-9}
Y-90	8.32×10^{-1}	5.55×10^{-16}	2×10^{-5}	2.8×10^{-11}
Sr-91	2.37	1.58×10^{-15}	5×10^{-5}	3.2×10^{-11}
Y-91	1.82×10^{1}	1.21×10^{-14}	3×10^{-5}	4×10^{-10}
Y-92	3.52×10^{-1}	2.35×10^{-16}	6×10^{-5}	3.9×10^{-12}
Mo-99	9.92×10^{3}	6.61×10^{-12}	4×10^{-5}	1.7×10^{-7}
I-131	5.52×10^{3}	3.68×10^{-12}	3×10^{-7}	1.2×10^{-5}
Te-132	5.60×10^{2}	3.67×10^{-13}	2×10^{-5}	1.8×10^{-8}
I-132	1.76×10^{2}	1.17×10^{-13}	8×10^{-6}	1.5×10^{-8}
I-133	3.35×10^{3}	2.23×10^{-12}	1×10^{-6}	2.2×10^{-6}
I-134	7.62	5.08×10^{-15}	2×10^{-5}	2.5×10^{-10}
I-135	1.88×10^{3}	1.25×10^{-12}	4×10^{-6}	3.1×10^{-7}
Cs-134	7.44×10^{2}	4.96×10^{-13}	9×10^{-6}	5.5×10^{-8}
Cs-136	7.10×10^{1}	4.73×10^{-14}	6×10^{-5}	7.9×10^{-10}
Cs-137	3.19×10^{3}	2.13×10^{-12}	2×10^{-5}	1.1×10^{-7}
Ba-140	1.88	1.25×10^{-15}	2×10^{-5}	6.3×10^{-11}
La-140	1.75	1.17×10^{-15}	2×10^{-5}	5.9×10^{-11}
Ce-144	6.85	4.57×10^{-15}	1×10^{-5}	4.6×10^{-10}
H-3	4.24×10^{9}	2.83×10^{-6}	3×10^{-3}	9.4×10^{-4}

μC = Micro Curies

Flow Rate In Circ. Water Disch. Canal = 15×10^{14} cc/yr

TABLE 2

Estimated Gaseous Radioactivity Release From
Indian Point Unit No. 2 and Resultant - Site Boundary Dose

Isotope	Estimated Annual Release Curies/yr	Integrated Whole Body Dose mr/yr	Fraction of MPC
Kr-85	5310	7	0.014
Xe-133	4540	6	0.012
Total	9850	13	0.026

$\frac{X}{Q} = 2.5 \times 10^{-5}$ sec/m^3

"Emergencies don't necessarily happen when the weather is fine and everybody is home listening to the telephone so that the question of back-up and organizational changes that are required because people aren't available or communication isn't just what it is expected to be, might be discussed in some detail.

The plan looks like a good one and it is quite elaborate if everything works out as it is expected to in that plan. But if it doesn't work out, what then happens?"

Answer:

The capability to respond effectively to emergencies at any time is necessary for the success of an emergency plan. This need is recognized in Applicant's Radiation Contingency Plan. Under that plan, persons immediately responsible for dealing with the emergency (i.e., the Contingency team) are part of the regular operating shift for the plant at the time of the emergency (see Page 2 of plan). In particular, the persons responsible for collecting the radiological, meteorological and other information needed to evaluate the emergency are the same persons assigned around-the-clock duties as licensed Reactor Operators, Senior Reactor Operators, and Health Physics Technicians. Accordingly, there is no need for home or outside telephone contact of Con Edison personnel immediately required for evaluation of and response to the emergency condition.

The communications available to reach the Atomic Energy Commission and the responsible agencies of the State of New York are provided through 24-hour manned telephone service. The governmental agencies in turn have available rosters of persons by name and home telephone who are available. In the

event of a telephone system breakdown at Indian Point, the station has available to it radio communications with the Con Edison System Operator who would then make contact with the individuals and State and Federal agencies involved.

In order to provide up-to-date rosters of telephone contacts, periodic reviews are performed to assure current information to those responsible for implementing the plan.

"But there have been two letters particularly from Advisory Committee on Reactor Safeguards, one of which was October 12, 1966, a letter addressed to the Chairman of the Atomic Energy Commission and its consists of some four pages outlining specific areas in which the Advisory Committee on Reactor Safeguards indicated that further research in Government should be undertaken.

Now, maybe it has been undertaken. We would be pleased to have both the Applicant and the Staff discuss that letter with specifics on fulfillment of the R&D that probably has been undertaken since 1966. We talk about a possible functional failure of the emergency core cooling system and other aspects of the entire operations. This does not apply solely to Con Edison. It applies to all reactors.

So I think this proceeding might give the Staff, specially, and the Applicant, if it could get the data, an opportunity to more or less update these areas of concern so that the record will show and the Advisory Committee will have an opportunity to review the transcript as to how the programs have been carried out."

Answer

The following are applicant's comments with respect to fulfillment of R&D undertaken since 1966 relative to the ACRS letters mentioned above.

1. On October 27, 1966, the Atomic Energy Commission appointed a task force to review the emergency core cooling systems and core protection. The task force issued a report in 1967 entitled: "Emergency Core Cooling - Report of Advisory Task Force on Power Reactor Emergency Cooling". Conclusion 12 and Appendices 7 and 8 of this report deal with the subject of molten mixtures of fuel, clad and other materials and means of handling them.

Additional studies dealing with molten mixtures of fuel, clad and other materials and means of handling them have been included in a broader program of review and evaluation of the

loss-of-coolant accident and the emergency core cooling experimental program started in 1968. The latest quarterly report from this program which is available to us (Reference 1) lists numerous technical papers and quarterly progress reports.

Design modifications were developed and incorporated in the Indian Point Unit No. 2 Emergency Core Cooling System which limit peak fuel clad temperature and restrict metal-water reaction so that continued effectiveness of the Emergency Core Cooling System is assured thus avoiding the onset of fuel clad melting. These modifications are increased capacity of the Emergency Core Cooling System by the addition of a pressurized accumulator to each coolant loop plus valving and piping changes which provide capability to maintain core cooling and containment cooling in the event of a passive failure in the safety injection system or service water system for the long term after a loss of coolant.

2. The PWR Full-Length Emergency Cooling Heat Transfer (FLECHT) Program was authorized by the AEC and performed by Westinghouse with objectives to obtain experimental flooding heat transfer data under simulated loss-of-coolant accident conditions for evaluating the heat transfer capabilities of pressurized water reactor emergency core cooling systems. The tests investigated the effects of peak power, power decay rate, maximum initial clad temperature, constant and variable flooding rates, inlet coolant subcooling, pressure,

flow blockage, borated coolant, and clad material. Bottom flooding heat transfer tests were conducted on full-length rod bundles. The tests studied transient heat transfer coefficients and clad temperatures, axial and radial pressure drop data, local coolant temperature and measurements of carry-over water.

The test results have verified the basic assumptions used in current reactor loss-of-coolant accident analysis, in particular, the effectiveness of bottom flooding and the importance of liquid entrainment as a heat transfer mechanism. A compilation of work performed under this program up to January 1970 has been published in References 2 through 5.

3. The development of practical systems for periodic inspection of Reactor Vessels was implemented by ANSI N45 Committee sponsored by the ASME as well as the Edison Electric Institute/TVA sponsored program carried out under contract to Southwest Research Institute. The efforts of the N45 Committee resulted in the formal adoption by the ASME in January, 1970 of Section XI as a practical method for inspection. The program sponsored by EEI/TVA relating to Reactor Vessel testing and inspection (RP79) was authorized in 1967 with completion of this program scheduled during 1972. A compilation of work performed under this program up to January 1, 1971 is listed in Reference 6.

4. A program recommended by the Pressure Vessel Research Committee was implemented in two parts. A very large program, known as the Heavy Section Steel Technology Program was funded by the AEC and administered by ORNL. This program undertook the developmental work necessary to improve the knowledge of fracture, leading to more positive assurance of pressure vessel safety. This program included very significant efforts in the development of fracture mechanics technology, test methods, analytical procedures, material property determinations, and crack propagation behavior. The latter also included radiation effects programs. It has been described in detail in References 7 through 9, and results have been published in semi-annual reports, References 10 through 16. The second part of the program, called the Industry Cooperative Program, undertook to evaluate better the properties of pressure vessel steels in heavy sections. Samples of many plates and forgings being used for Reactor Pressure Vessels were tested to determine the variation in properties that could be expected from place to place and through the thickness of actual production material. Another aspect of this program was to evaluate the adequacy of the non-destructive examination procedures used to assure the integrity of vessels. Plate, forgings, castings and welds are included in this study. Results of this work are included in the reports of progress of the Welding Research Council. Reference 17 lists the results of numerous papers.

Extensive work on fracture analysis and irradiation effects has also been accomplished at the Naval Research Laboratory funded by HSST and other AEC programs. Pertinent papers are listed in References 18 through 24. Westinghouse and The Empire State Atomic Development Associates (ESADA) have also participated in this work; results of these efforts have been technically incorporated into the HSST reports.

Additional efforts to understand fracture have been proceeding at Universities, such as Lehigh and the Colorado School of Mines. Private industry has also carried out extensive work with its own funds, publishing its results in various symposia and in conjunction with HSST Program. Further References 25 through 34 are given wherein results of this large body of work are reported. Some of these references contain many technical papers dealing directly with the results of the work discussed above.

Generally, it has been found that the effect of thickness can readily be handled by either transition temperature or fracture mechanics approaches. Variations in properties have been found to be well within the allowances made for them. Crack propagation rate studies have shown that present criteria are very conservative.

The extensive work on assessment of radiation effects has also shown that the changes in properties can be accurately predicted and will not be detrimental. Several different approaches to setting safe operating procedures have been developed. These all result in similar criteria, giving additional confidence that the subject of fracture is well understood, and therefore can be prevented.

5. The hypothetical large reactivity insertion accident which is postulated for Pressurized Water Reactors is the Rod Ejection Accident. This postulated event, in which a rod control cluster is rapidly ejected from the reactor core, would cause a strong reactivity insertion and a dynamic power redistribution.

As is shown in the FSAR, the hypothetical ejection of a control cluster from the "full power" configuration or from the "zero power" configuration does not lead to overpressurization of the primary system or to gross fuel melting. This FSAR analysis relies on methods which agree with the appropriate Spert experiments. Analytical techniques described in Reference 35 show that the methods used in the Rod Ejection analysis are conservative in comparison with more recently developed spatial kinetic transient analyses for cases of interest.

Westinghouse has developed space-dependent kinetic computational capability. Methods are now in use which treat three space dimensions, two neutron energy groups, fuel temperature "local" feedback, and transient water density "local" feedback. Sufficient spatial detail is now practical to allow detailed analyses of such events as Rod Ejection. Other effects which are of special interest, in certain unusual cases, are also modeled directly; e.g., non-unity importance of delayed neutrons, and spatially varying delayed neutron yield from fission. These methods are related to the WIGL codes (Reference 36) although separately developed, and give accuracy/stability performance equal to that of the W-R method, Reference 37. These advanced methods, by allowing direct engineering analysis of transients in which spatial effects are important, reduce the uncertainty in the study of reactivity accidents. In addition, the Westinghouse spatial kinetic models are direct extensions of the Westinghouse spatial static models which have been confirmed by reactor operating experience and are reconfirmed in each reactor by startup physics tests and periodically-repeated physics tests.

Experimental results from the planned PBF test program, as to fuel failure modes and fuel failure propagation modes (if any), while of definite interest, are not essential to the design and safe operation of Westinghouse PWR's. Even when

subjected to hypothetical Rod Ejection reactivity insertions from full power to zero power, the PWR is inherently limited by conservative design to staying below the point of gross fuel melting or significant primary pressure surges.

6. Stronger steels were not required for the Indian Point Unit No. 2 reactor vessel. The materials used (see FSAR Section 4) were those for which a wealth of data and exprience already existed.

7. In 1914, the American Society of Mechanical Engineers first published its boiler code for use by industry. Along with this first code, various ASME sub-committees were established to remain abreast of industrial experience and developments in several areas including design, fabrication, inspection and testing. Members of these sub-committees are selected from both industry and government (i.e., Insurance Companies, Manufacturers, Utilities, Designers, Regulatory Bodies) with careful attention given to a proper balance of representation amongst the participants. Each subcommittee member is, of course, active in his particular field. This results in aggressive action by all committees to maintain the particular segment of the code as current and viable as possible. Recommendations are made by individual members on a continuing basis as advances in technology are made or as other specific requirements arise. As a result, commercial experience on both

nuclear and non-nuclear components is considered on a current basis with applicable codes and regulations being updated frequently. The nuclear codes are an outgrowth of these recommendations and review work, recognizing the additional requirements of public safety introduced by the advent of commercial nuclear power.

8. The Plutonium Recycle Program was initiated in 1964 to ensure the design, fabrication, and operation of plutonium recycle fuel on a commercial basis as well as identifying safety-related problems associated within this type of fuel. The program includes four years of operation and post-irradiation examination of Saxton plutonium fuel, two joint projects with the Edison Electric Institute, criticality studies for Empire State Atomic Development Associates (ESADA) and operation of the Westinghouse Plutonium Fuels Development Laboratory (PFDL) which was completed in 1969. In addition, design and fabrication development work is being conducted by Westinghouse. This includes fabrication of plutonium fuel reload assemblies in the development laboratory during 1971-1974, plus the design and development work leading to construction of a commercial plutonium fabrication plant during this period.

The first part, the Saxton Plutonium Program, was a joint effort with the AEC and the Saxton Nuclear Experimental Corporation from its inception in 1964 until mid-1970 after which it was supported completely by Westinghouse. This part of the program has two basic objectives: to demonstrate performance of mixed oxide fuel at linear power and burnup levels consistent with modern PWR technology; and to obtain design information on depletion and transuranic isotope generation characteristics of plutonium fuel at higher burnup.

An extensive Core II post-irradiation program was completed in early 1970. The peak burnup evaluated was 29,000 megawatt days per tonne (MWd/t). Saxton has continued to operate satisfactorily since starting Core III power operations in December 1969. Since then, a significant number of plutonium rods have operated at 19 kW/ft. Burnup for this fuel has achieved 44,000 MWd/t, demonstrating the peak power and burnup levels required for modern PWR's.

The EEI-Westinghouse Plutonium Utilization Study combined with the ESADA-Westinghouse Critical Experiments (1966-68) was the next part of the program. This EEI Project RP72 Phase I program was primarily an analytical feasibility study of the technical and economic parameters influencing the use of plutonium in a PWR.

The EEI-Westinghouse Plutonium Recycle Demonstration Program was undertaken (EEI Project RP-72) under support of the AEC, Contract (30-1)-4167, to license, operate and evaluate a representative number of plutonium fuel rods in the San Onofre reactor. This demonstration experience under actual utility operating conditions will complement the material and design recycle information being generated in the Saxton test reactor, AEC Contract AT(30-1)-3385. In order to achieve a meaningful demonstration, Westinghouse first determined the characteristics of future commercial recycle fuel, based on technical and economic comparisons of various recycle methods for the Indian Point Unit No. 2 design. The comparisons included core power distribution calculations, plus an in-depth analysis of reactivity coefficients and control requirements.

Four demonstration assemblies were loaded in the San Onofre reactor in October and the reactor returned to full power in November 1970. Currently, these assemblies have achieved a burnup of 3,000 MWd/t. On site tests and inspections at San Onofre, plus post-irradiation examinations by W, are planned starting in 1972. Based on the results of these demonstration programs, no safety problems are anticipated for plutonium recycle in pressurized water reactor plants. Progress of the program through December 1970 is described in Reference 38.

9. Methods for detecting leakage in primary coolant systems of reactors already existed at the time of the October 12, 1966 ACRS letter. However, as a result of work performed by Westinghouse in 1966, an additional method (described in Supplement 7, PSAR) determines total leakage of steam and water systems inside containment by measuring the condensate collected by the containment cooling coils. Several other W Pressurized Water Reactor Plants now utilize this method. With this method, Indian Point Unit No. 2 has four methods available for early detection of small leaks. These methods with sensitivities and basis for design are described in Section 6.7 of the FSAR.

10. Studies of dilution, dispersion and transport of liquid radioactive wastes have been conducted for the Hudson River and are reported in the Indian Point Unit No. 2 FSAR, Section 2.5. Other such studies, necessarily site-dependent, have also been conducted by others for other plants as part of normal efforts to obtain AEC regulatory staff licensing approval.

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