ATTACHMENT

Proposed Storage of Indian Point Unit No. 2 Spent Fuel In the Indian Point Unit No. 1 Fuel Handling Building

Consolidated Edison Company of New York, Inc.
Indian Point Unit Nos. 1 & 2
Docket Nos. 50-3 & 50-247

May, 1978

TABLE OF CONTENTS

SUBJECT	Page Numbers
1. Introduction	A-1
2. Description of Indian Point Unit No. 1 Fuel Storage Facilities.	A-3
3. Description of Proposed Modification Including Design Concepts and Design Criteria, where applicable.	A-5

1.0 Introduction

Consolidated Edison Company of New York, Inc. (Con Edison) owns two nuclear power plants, Indian Point Unit Nos. 1 and 2, located at Buchanan, New York. Indian Point Unit No. 2 (IP2) is an operating 2758 MWth Pressurized Water Reactor (PWR). Indian Point Unit No. 1 (IP1), a 615 MWth PWR, was shut down on October 31, 1974 and is presently in a defueled condition. Fig. A-1 shows the general layout of the site including location of the IP1 and IP2 fuel storage buildings. A third plant at the site, Indian Point Unit No. 3, is owned and operated by the Power Authority of the State of New York.

In December of 1975, the NRC amended license DPR-26 to permit modification of the IP2 spent fuel racks to provide additional storage capacity. These modifications increased spent fuel storage capacity from 264 assemblies (approximately 4 regions) to 482 assemblies (approximately 7½ regions). Table A-1 presents the current estimated schedule of spent fuel discharges from IP2. Region 4 will be the last region that can be discharged and stored in the pool while still maintaining full core reserve (FCR). Consequently, additional spent fuel storage will be required in time for the discharge of the 5th region, which is estimated to occur in 1982.

A feasibility study has indicated that a portion of the IP1 spent fuel facility could provide sufficient storage for twelve regions of IP2 spent fuel. This capacity to store IP2 spent fuel would not interfere with a return to service of IP1. Con Edison therefore proposes to modify the East Storage Pool of IP1 to store an estimated 800 IP2 spent fuel assemblies. This report sets forth the design concepts and design criteria to accomplish this modification.

2.0 Description of Indian Point Unit No. 1 Fuel Storage Facilities (see Figure A-2)

The IP1 Fuel Handling Building has seven (7) pools. These pools, which are lined with a Carboline four coat phenolic system, have the following approximate water capacities:

(1)	Water Storage Pool	272,000 gallons
(2)	East Storage Pool	169,000 gallons
(3)	Fuel Transfer Pool	87,000 gallons
(4)	West Storage Pool	200,000 gallons
(5)	Failed Fuel & Aux. Pool	48,000 gallons
(6)	Disassembly Pool	48,000 gallons
(7)	Cask Load Pool	48,000 gallons

At the present time the West Storage Pool contains 160 IP1 assemblies and Core B filler rods. The East Storage Pool, which presently contains miscellaneous IP1 Core A components, is the pool Con Edison is proposing to use for storage of IP2 spent fuel. It is planned to move IP2 spent fuel to IP1 by truck.

2.1 Physical Description

Con Edison proposes to modify the East Storage Pool of the IP1 Fuel Handling Building to store IP2 spent fuel. The IP2 spent fuel will be transferred via spent fuel casks from the IP2 pool to the IP1 Fuel Handling Building, a distance of approximately 450 ft. The cask load pool will be used to unload the IP2 spent fuel assemblies and transfer them underwater, to the East Storage Pool. A description of pertinent equipment, including the cooling system, follows.

2.1.1 Cooling System

Cooling can be accomplished by use of any of three submerged suction centrifugal type pumps located in the water storage pool pump pit: two spent fuel cooling water pumps and one spent fuel cooling water transfer pump.

The pump discharge can be directed through either of two shell and tube type heat exchangers. The cooled heat exchanger discharge may then be directed to the East Storage Pool or any of the remaining pools.

2.1.2 Pool Gates

Except for the Water Storage Pool, the various pools in the Fuel Handling Building are all interconnected by means of double gates. By use of these gates, adjacent pools can be isolated or grouped as desired.

2.1.3 Pool Purification

Purification of the water in the Fuel Handling Building storage pools is accomplished by directing a portion of the pump discharge either through or bypassing the heat exchanger through the spent fuel filters and/or spent fuel mixed bed ion exchangers, and then back into one or more of the pools.

3.0 Description of Proposed Modifications Including Design Concepts and Design Criteria, where applicable.

This section will present the conceptual design of the proposed modifications and will include information on rack design and spacing, burnup and cooling assumptions for the spent fuel, adequacy of present pools, technical specifications changes and applicable criteria and codes. Preliminary criticality, heat removal and radiological calculation have indicated that as many as 800 assemblies can be stored in the East Storage Pool of IP1, although Con Edison has not decided exactly how many IP2 spent fuel assemblies will be stored there. Con Edison envisions, based upon scoping studies, that the center-to-center spacing of fuel assemblies in the new racks will be in the range of approximately 10 - 12 inches, depending upon how many assemblies are to be stored. Appropriate criticality criteria will be met. All appropriate accident analyses and shielding considerations will be provided in our final design/licensing submittal, presently scheduled for April, 1979.

The spent fuel racks will meet the design criteria of the appropriate ANSI standard. The racks will be designed to resist the combined loadings of the dead weight of the cells, the weight of the spent fuel assemblies and seismic loads (seismic Class I). The rack design is not available at this time but will be a part of the final design submittal. The existing pools have been analyzed and have been found capable of withstanding the effects of the site Safe Shutdown Earthquake of 0.15g. Con Edison is investigating the use of certain neutron absorbing materials (poisons) in its design. The final licensing submittal will describe which poison, if any, will be used.

IP2 spent fuel will be initially stored in the IP2 spent fuel pool before being transferred to the IP1 pool. The final submittal will take credit for time spent fuel is cooled in the IP2 spent fuel pool (at least one year). In addition, credit may be taken for at least one cycle of burnup (conservative assumptions on minimum assembly burnup would be made). Appropriate administrative controls to meet these assumptions will be developed and described in the final licensing submittal. Preliminary calculations have shown that under normal conditions, the present cooling system is adequate to meet the heat load of the maximum number of IP2 assemblies that will be stored at IP1. Although the present cooling system is not Seismic Class I, it is shown below that following a postulated earthquake, there would be sufficient time to furnish alternate supplies of cooling water to the pool.

In view of the fact that IP2 fuel to be stored in IP1 will already have had an estimated cooling time of at least one year, decay heat generation is relatively small and well within the cooling capability of one of the existing IP1 spent fuel heat exchangers. Should this cooling be lost, water heatup rates in the East Storage Pool (neglecting any water volume which may be contained in other pools) are approximately 3-4^{o}F per hour. Should bulk boiling occur, water level would drop at a rate of less than 2 inches per hour. Ample time would thus be available to provide makeup water from alternate sources.

IP1 spent fuel pool cleanup systems are of sufficient design capacity to maintain necessary water clarity and purity to minimize occupational exposures when fuel storage activities are being conducted in IP1. The IP1 fuel pool cleanup system includes both filters (4) and mixed bed demineralizers (2) to remove both suspended solids and dissolved ionized solids and gases.

Because of the estimated minimum decay time of one year for IP2 spent fuel before being stored in IP1 facilities, the only isotopes which must be considred in an anlysis of possible radiological consequences of fuel handling accidents or failures are the isotopes I-129 and Kr-85. All other noble gases and iodines are not present in any significant quantity after one year cooling. Release of the fuel-clad gap fractions (0.3) of I-129 and Kr-85 from all assemblies stored in the pool (about 800) could result in a gamma whole body dose of less than 1 rem and a thyroid dose (assuming no pool Decontamination Factor -DF) less than 10% of 10 CFR100 limits.

A list of applicable codes, standards, specifications, etc. that will be used in any phase of this project will be included in Con Edison's final submittal. Technical Specifications changes will also be included in the final submittal.

Table A-1

CONSOLIDATED EDISON CO. OF NEW YORK, INC.

INDIAN POINT UNIT NO. 2

SPENT FUEL DISCHARGES

Calender Year	Estimated Annual Discharges MTU	No. of Assys.	Region No.
Current Inventory	32.6	72	1
1978	27.1	60	2
1979	29.0	64	3
1980	----	--	-
1981	29.0	64	4
1982	29.0	64	5
1983	29.0	64	6
1984	----	--	--
1985	29.0	64	7
1986	29.0	64	8
1987	29.0	64	9
1988	----	--	-
1989	29.0	64	10
1990	29.0	64	11
1991	29.0	64	12

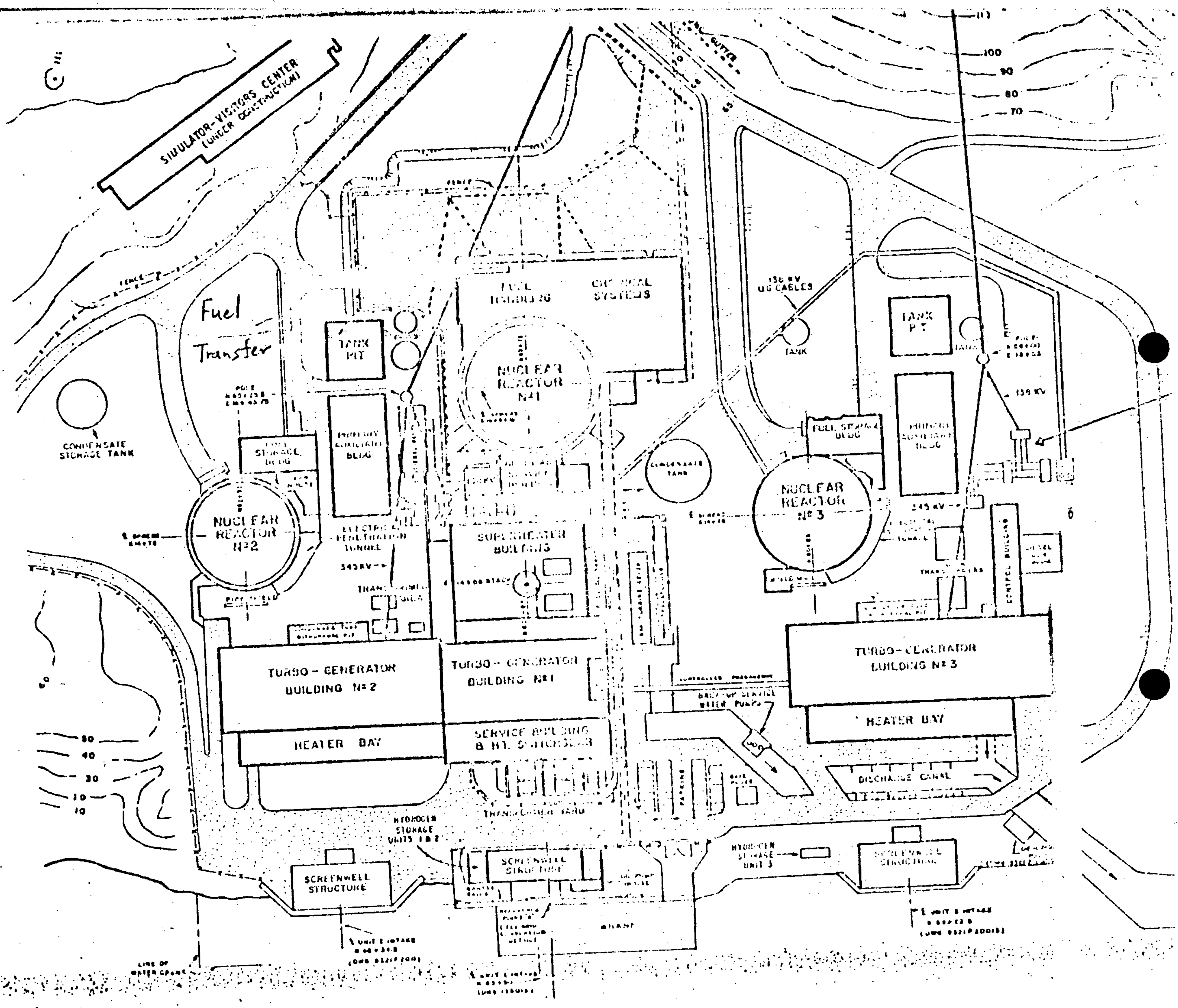
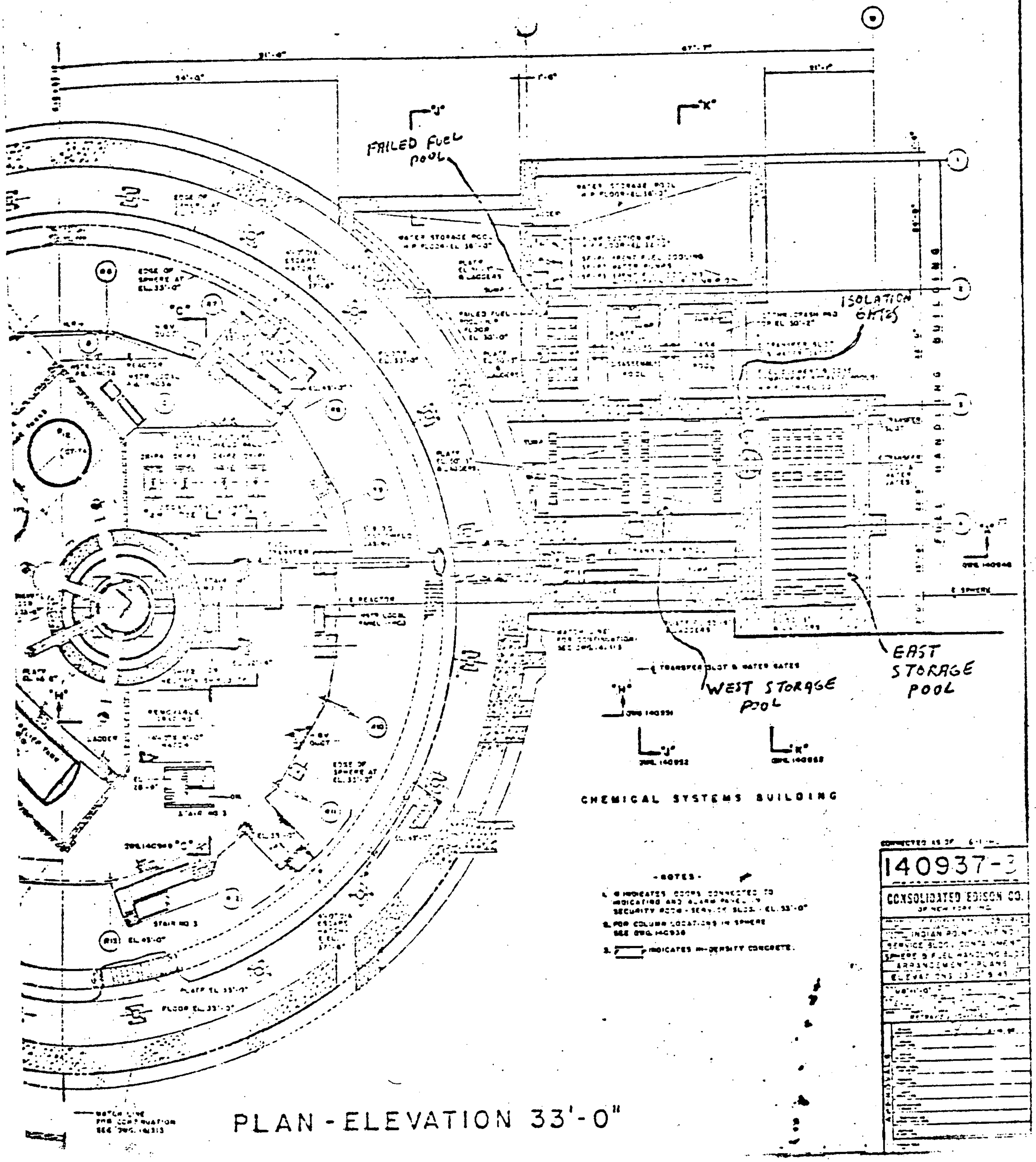

SIMULATOR-VISITORS CENTER
(UNDER CONSTRUCTION)
FENCE
Fuel
Transfer
TANK PIT
FUEL HANDLING
CHEMICAL SYSTEMS
NUCLEAR REACTOR Nº1
CONDENSATE STORAGE TANK
FUEL STORAGE BLDG
PRIMARY AUXILIARY BLDG
NUCLEAR REACTOR Nº2
ELECTRICAL PENETRATION TUNNEL
345KV
TRANSFORMERS
SUPERHEATER BUILDING
CONDENSATE TANK
138 KV UG CABLES
TANK
NUCLEAR REACTOR Nº3
138 KV
345 KV
CONTROL BUILDING
TURBO-GENERATOR BUILDING Nº2
TURBO-GENERATOR BUILDING Nº1
TURBO-GENERATOR BUILDING Nº3
HEATER BAY
SERVICE BUILDING & HT. SWITCHGEAR
HEATER BAY
BACK-UP SERVICE WATER PUMPS
CONTROLLED PASSAGEWAY
DISCHARGE CANAL
PARKING
TRANSFORMER YARD
HYDROGEN STORAGE UNITS 1 & 2
HYDROGEN STORAGE UNIT 3
SCREENWELL STRUCTURE
SCREENWELL STRUCTURE
SCREENWELL STRUCTURE
LINE OF WATER EDGE
100
90
80
70
65
60
50
40
30
20
10

FIGURE A-1

SPENT FUEL POOLS

Figure A-2



PLAN - ELEVATION 33'-0"

William J. Cahill, Jr.
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N Y 10003
Telephone (212) 460-3819

March 29, 1978

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Reid:

The information requested by your letter of May 18, 1977 regarding the Indian Point Unit No. 2 Reactor Vessel Material Surveillance Program is forwarded herewith as Attachment A to this letter. This submittal supersedes our partial response of July 26, 1977. Should you or your staff have any further questions, we will be pleased to discuss them with you. Forty (40) copies of WCAP-7323, "Indian Point Unit No. 2 Reactor Vessel Radiation Surveillance Program," which describes the reactor vessel material surveillance program is provided as attachment B to this letter.

Very truly yours,

William J. Cahill, Jr

William J. Cahill, Jr
Vice President

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TO FILES

INDIAN POINT UNIT NO. 2 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

1.) The estimated maximum fluence (E > 1 Mev) at the inner surface of the reactor vessel wall as of March 31, 1977 is 8.51×10^{17} n/cm^2.

2.) The effective full power years (EFPY) of operation accumulated as of March 31, 1977 is 1.74 EFPY.

3.) Fabrication of the reactor vessel was performed by Combustion Engineering, Inc.

4.) a.) Sketch of the reactor vessel showing all materials in the beltline region is shown in Figure 1.

b.) Information on each of the welds in the beltline region is shown in Tables 1 through 4.

c.) Information on each of the plates in the beltline region is shown in Tables 4 through 7.

5.) Information relative to weld and plate material included in the material surfeillance program is shown in Tables 1 through 3 and 5 through 7.

FIGURE 1

IDENTIFICATION AND LOCATION OF BELTLINE REGION
MATERIAL FOR THE INDIAN POINT UNIT NO. 2 REACTOR VESSEL

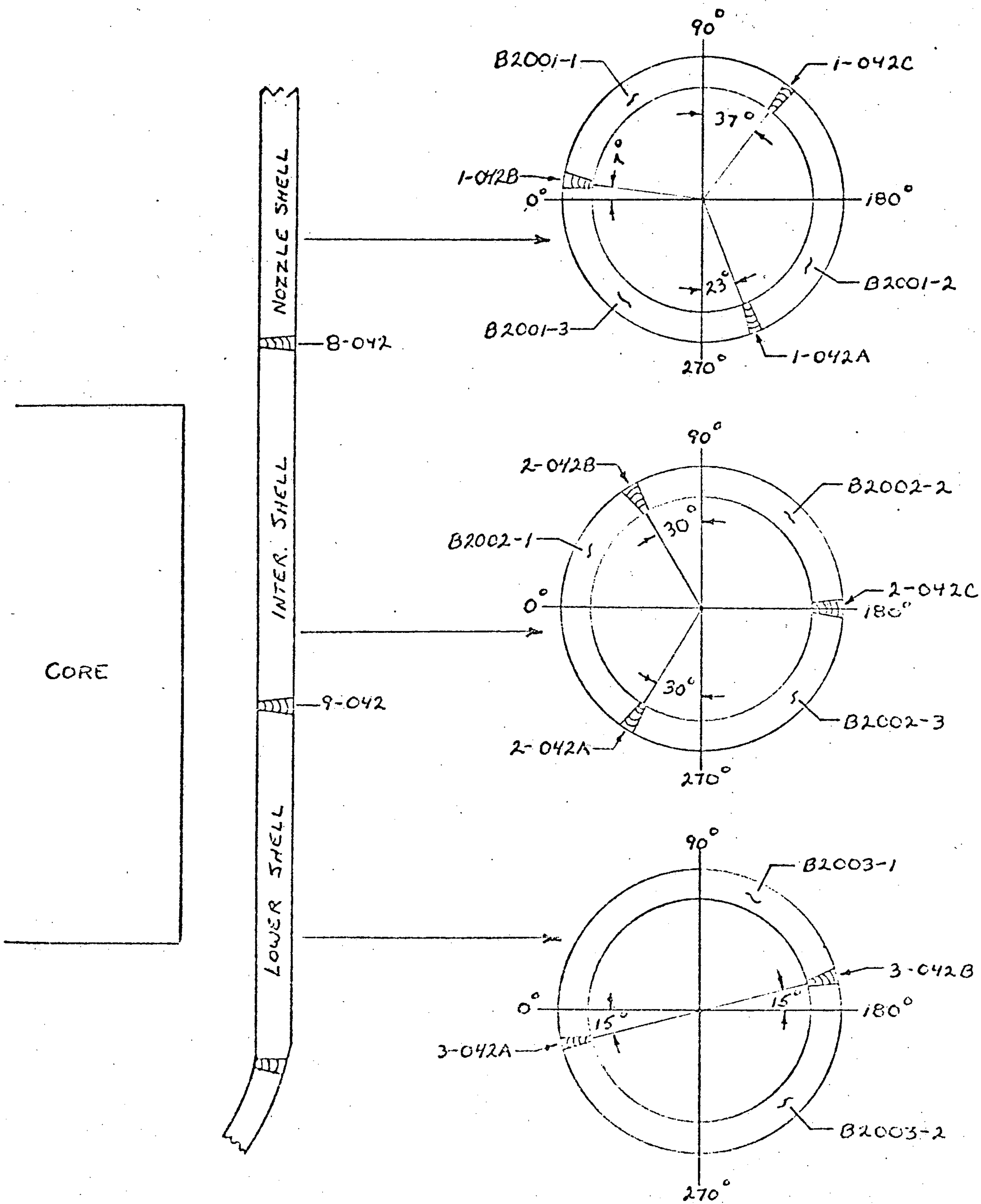

TABLE 1

IDENTIFICATION OF INDIAN POINT UNIT NO. 2 REACTOR VESSEL BELTLINE REGION WELD METAL

Weld Location	Welding Process	Weld Control No.	Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	Post Weld Heat Treatment
Nozzle Shell Vertical Seam 1-042 A, B&C	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 $\pm$ 25°F-25 HR-FC
Nozzle Shell to Inter. Shell Circle Seam 8-042	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 $\pm$ 25°F-25 HR-FC
Inter. Shell Vertical Seam 2-042 A, B&C	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 $\pm$ 25°F-25 HR-FC
Inter. Shell to Lower Shell Circle Seam 9-042	Submerged Arc	M1.03	RACO 3 +Ni 200	34B009 N9867A	Linde 1092	3708	1150° $\pm$ 25°F-40 HR-FC
Lower Shell Vertical Seams 3-042 A&B	Submerged Arc	-	RACO 3 +Ni 200	W5214 -	Linde 1092	3576	1150 $\pm$ 25°F-40 HR-FC
Surveillance Weld	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1150 $\pm$ 25°F-19 3/4 HR-FC

Table 2

Chemical Composition of Vessel Beltline Region Weld Metal

Weld Wire		Flux		Weight Percent								
Type	Heat No.	Type	Lot No.	C	Mn	P	S	Si	Mo	Cr	Ni	Cu
RACO 3	W5214	Linde 1092	3600	.11	1.20	.021	.012	.19	.52	--	--	--
RACO 3	348009	Linde 1092	3708	.14	2.01	.010	.017	.04	.51	--	--	-- *
RACO 3	W5214	Linde 1092	3576	.12	1.15	.021	.012	.21	.56	--	--	--

Surveillance Weld - Not Performed

*Chemical analysis of bare wire - No as deposited analysis is available

Table 3

Mechanical Properties of Vessel Beltline Region Weld Metal

Weld Wire		Flux									
Type	Heat No.	Type	Lot No.	T_{NDT}* °F	Energy at 10°F Ft-Lbs	RT_{NDT}* °F	Shelf Energy Ft-Lbs	YS KSI	UTS KSI	Elong %	RA %
RACO 3	W5214	Linde 1092	3600	0	103,93,95	0	--	65.5	80.0	31.0	71.5
RACO 3	348009	Linde 1092	3708	0	84,71,90	0	--	67.9	84.2	31.0	69.8
RACO 3	W5214	Linde 1092	3576	0	57,51,69	0	--	68.5	85.0	27.5	68.5
Surveillance Weld				0	78,74,81	0	121	64.75	80.85	27.7	72.7

*Estimated per NRC Standard Review Plan Section 5.3.2

Table 4

Maximum End-of-Life Fluence at Vessel Inner Wall Locations

Plate or Weld Location	Seam or Plate No.	Fluence N/CM^2
Nozzle Shell Vertical Seam	1-042A	6.6×10^{17}
" " " "	1-042B	4.4×10^{17}
" " " "	1-042C	1.1×10^{18}
Nozzle Shell to Inter. Shell Circle Seam	8-042	1.3×10^{18}
Inter. Shell Vertical Seam	2-042A	8.8×10^{18}
" " " "	2-042B	"
" " " "	2-042C	5.0×10^{18}
Inter. Shell to Lower Shell Circle Seam	9-042	1.6×10^{19}
Lower Shell Vertical Seam	3-042A	7.0×10^{18}
" " " "	3-042B	"
Nozzle Shell Plate	B2001-1	1.3×10^{18}
" " "	B2001-2	"
" " "	B2001-3	"
Inter. Shell Plate	B2002-1	1.6×10^{19}
" " "	B2002-2	"
" " "	B2002-3	"
Lower Shell Plate	B2003-1	"
" " "	B2003-2	"

Table 5

Identification of Reactor Vessel Beltline Region Plate Material

Component	Plate No.	Heat No.	Mat'l Spec No.	Supplier	Heat Treatment: Austenitize	Heat Treatment: Temper	Heat Treatment: Stress Relief
Nozzle Shell	B2001-1	B4679	A302B Mod.	Lukens	1550-1650°F-4Hrs-WQ	1200-1250°F-4Hr-AC	1125-1175°F-60Hr-FC
" "	B2001-2	B4701	"	"	"	"	"
" "	B2001-3	A9870	"	"	"	"	1125-1175°F-50Hr-FC
Inter Shell	B2002-1	B4688	"	"	"	"	"
" "	B2002-2	* B4701	"	"	"	"	"
" "	B2002-3	B4922	"	"	"	"	1125-1175°F-40Hr-FC
Lower Shell	B2003-1	B4791	"	"	"	"	"
" "	B2003-2	B4782	"	"	"	"	"

*Surveillance Material

Table 6

Chemical Composition of Reactor Vessel Beltline Region Plate Material

Weight Percent

Plate No.	C	Mn	P	S	Si	Ni	Mo	Cu
B2001-1	.22	1.35	.010	.022	.24	.50	.46	.20
B2001-2	.23	1.27	.011	.021	.23	.43	.47	.14
B2001-3	.23	1.35	.012	.025	.26	.50	.48	.19
B2002-1	.20	1.28	.010	.019	.25	.58	.46	.25
B2002-2 *	.22	1.30	.014	.020	.22	.46	.50	.14
B2002-3	.22	1.29	.011	.018	.25	.57	.46	.14
B2003-1	.23	1.33	.011	.025	.23	.66	.48	.20
B2003-2	.21	1.30	.010	.021	.23	.48	.45	.19

*Surveillance Material - No analysis performed other than reported by supplier.

Table 7

Mechanical Properties of Reactor Vessel Beltline Region Plate Material

Plate No.	T_{NDT} °F	RT_{NDT}* °F	Shelf Energy* Ft-Lb	YS KSI	UTS KSI	Elong %	RA %	
B2001-1	-10	24	69	67.25	87.75	26.00	64.45	
B2001-2	-10	18	63.5	63.25	85.25	27.25	65.75	
B2001-3	-10	25	69	65.25	86.75	25.00	63.75	
B2002-1	-20	34	70	70.75	91.50	25.00	64.75	
B2002-2	-30	21	73	65.00	85.25	26.50	67.00	
B2002-3	-10	21	73.5	68.95	90.50	26.75	67.75	
B2003-1	-20	20	71	65.75	87.25	27.75	65.50	
B2003-2	-20	-20	88	61.25	81.60	30.75	70.50	
B2002-1	-	34	76	67.17	88.40	25.20	67.6	Surveillance Test Data
B2002-2	-	34	75	64.55	87.15	27.65	69.8	Surveillance Test Data
B2002-3	-	39	72.5	65.32	87.32	26.30	67.0	Surveillance Test Data

*Estimated from longitudinal data per NRC Standard Review Plan Section 5.3.2

Attachment B

WCAP-7323

Indian Point Unit No. 2 Reactor Vessel
Radiation Surveillance Program