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SUBJECT: Forwards Rev 14 to "FSAR," which include approved & implemented changes re Thermal Uprate Project. See UFSAR

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L-97-59
10 CFR 50.4
10 CFR 50.71

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Updated Final Safety Analysis Report Revision 14

Florida Power and Light Company has completed Revision 14 of the Turkey Point Units 3 and 4 Final Safety Analysis Report (FSAR). As specified in 10 CFR 50.4(b)(6), ten additional copies of the revision are enclosed. Please note that a separate complete set of FSAR-related plant drawings are also provided for each copy of the FSAR. This set of drawings addresses an NRC concern for the quality (clarity) of the drawings previously submitted.

This special revision includes only the approved and implemented changes directly related to the Thermal Uprate Project. Please note that this special revision is not intended to satisfy the submittal requirements of 10 CFR 50.71(e)(4); that revision will be submitted within six months after the end of the next Unit 4 refueling outage, or approximately late May, 1998.

Very truly yours,

R. J. Hovey
Vice President
Turkey Point Plant

JEK

Attachment

cc: L. A. Reyes, Regional Administrator, Region II, USNRC
T. P. Johnson, Senior Resident Inspector, USNRC, Turkey Point

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50-250 FP&L TURKEY POINT 3

FIGURE AND ENGINEERING DRAWING
CROSS-REFERENCES

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<<NUDOCS/AD>> Nuclear Regulatory Commission ADQ42 V6.3.23.0
==== TCON64 ========== Accession Number - 9705300274 ====== Start ==== End ===
Availability: PDR Format: * Microfilm Address: 93179-001 93183-056
Size: 1p.

Document Type: Incoming Correspondence Issued: 970512
Desc/: Forwards Rev 14 to "FSAR," which include approved & implemented
Title: changes re Thermal Uprate Project.
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Authors: HOVEY,R.J. Florida Power & Light Co.

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Dockets: 05000250 50-250 Turkey Point Plant, Unit 3, Florida Power and Light C
05000251 50-251 Turkey Point Plant, Unit 4, Florida Power and Light C
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FSAR UPDATE REVISION 14
FIGURE AND ENGR'G DRAWING CROSS-REFERENCES
(FOR DISTRIBUTION WITH FSAR UPDATE)

ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-A-60	1	11	9.6A-8	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION FIRE ZONES AND BARRIERS FLOOR PLAN EL. 10'-0"	13
5610-A-60	2	2	9.6A-12	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION DETECTION, SUPPRESSION, & LIGHTING FLOOR PLAN EL. 10'-0"	13
5610-A-61	1	14	9.6A-9	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION FIRE ZONES AND BARRIERS FLOOR PLAN EL. 18'-0"	13
5610-A-61	2	6	9.6A-13	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION DETECTION, SUPPRESSION, & LIGHTING FLOOR PLAN EL. 18'-0"	13
5610-A-62	1	9	9.6A-10	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION FIRE ZONES AND BARRIERS FLOOR PLAN EL. 30'-0"	13
5610-A-62	2	4	9.6A-14	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION DETECTION, SUPPRESSION, & LIGHTING FLOOR PLAN 30'-0"	13
5610-A-63	1	10	9.6A-11	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION FIRE ZONES AND BARRIERS FLOOR PLAN EL. 42'-0"	13
5610-A-63	2	7	9.6A-15	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION DETECTION, SUPPRESSION, & LIGHTING FLOOR PLAN EL. 42'-0"	13
5610-C-2		24	1.2-1	TURKEY POINT PLANT UNITS 3 & 4 GENERAL BUILDING ARRANGEMENT PLAN	13

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FIGURE AND ENGR'G DRAWING CROSS-REFERENCES
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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-C-2		24	11.2-3	TURKEY POINT PLANT UNITS 3 & 4 GENERAL STATION AREA	13
5610-C-1168	2	4	2.2-3	TURKEY POINT PLANT UNITS 3 & 4 GENERAL SITE FEATURES	13
5610-C-1695	1	3	5G-1	TURKEY POINT PLANT UNITS 3 & 4 EXTERNAL FLOOD PROTECTION FLOOD PROTECTION BARRIERS PLANT ARRANGEMENT	13
5610-C-1695	2	0	5G-2	TURKEY POINT PLANT UNITS 3 & 4 EXTERNAL FLOOD PROTECTION PERIMETER FLOOD WALL DETAILS	13
5613-E-11	1	10	8.2-4a	TURKEY POINT PLANT UNIT 3 ELECTRICAL 125V DC AND 120V INSTRUMENT AC ON LINE DIAGRAM - SHEET 1	13
5613-E-11	2	9	8.2-4b	TURKEY POINT PLANT UNIT 3 ELECTRICAL 125V DC AND 120V INSTRUMENT AC ONE LINE DIAGRAM - SHEET 2	13
5614-E-11	1	7	8.2-4d	TURKEY POINT PLANT UNIT 4 ELECTRICAL 125V DC AND 120V INSTRUMENT AC ONE LINE DIAGRAM - SHEET 1	13
5614-E-11	2	12	8.2-4e	TURKEY POINT PLANT UNIT 4 ELECTRICAL 125V DC AND 120V INSTRUMENT AC ONE LINE DIAGRAM - SHEET 2	13
5613-E-12		4	8.2-4c	TURKEY POINT PLANT UNIT 3 ELECTRICAL 125V DC AND 120V INSTRUMENT AC ONE LINE DIAGRAM - SHEET 3	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-E-12		5	8.2-4f	TURKEY POINT PLANT UNIT 4 ELECTRICAL 125V DC AND 120V INSTRUMENT AC ONE LINE DIAGRAM - SHEET 3	13
5610-E-54-1	1	10	8.2-6	TURKEY POINT PLANT UNITS 3 & 4 COMPOSITE DRAWING OF CONTAINMENT ELECTRICAL PENETRATION CANISTERS	13
5610-E-54A-1		3	8.2-7	TURKEY POINT PLANT UNITS 3 & 4 5KV ELECTRICAL POWER PENETRATION ASSEMBLY	13
5610-M-51		3	11.2-2	TURKEY POINT PLANT UNITS 3 & 4 AREA RADIATION ZONE PLAN FULL POWER OPERATION WITH 1% FAILED FUEL	13
5610-M-55		6	1.2-2	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT PLAN EL. 10'-0"	13
5610-M-56		36	1.2-3	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT GROUND FLOOR PLAN EL. 18' -0"	13
5610-M-56		36	11.2-1	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT GROUND FLOOR PLAN EL. 18'-0"	13
5610-M-57	1	16	1.2-4	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT OPERATING FLOOR PLAN EL. 42'-0" & EL 58'-0"	13
5610-M-58		10	1.2-5	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT MEZZANINE FLOOR PLAN AND SECTION "A - A"	13

ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-M-59		9	1.2-6	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT SECTIONS "B - B" AND "C - C"	13
5610-M-60		5	1.2-7	TURKEY POINT PLANT UNITS 3 & 4 GENERAL ARRANGEMENT SECTIONS "D - D" & "E - E"	13
5610-M-63		10	7.7-1	TURKEY POINT PLANT UNITS 3 & 4 CONTROL ROOM EQUIPMENT LOCATIONS	13
5610-M-85	1	5	9.9-3 SH 1	TURKEY POINT PLANT UNITS 3 & 4 DC EQUIPMENT/INVERTER ROOMS HVAC SHEET 1	13
5610-M-85	2	2	9.9-3 SH 2	TURKEY POINT PLANT UNITS 3 & 4 DC EQUIPMENT/INVERTER ROOMS HVAC SECTIONS SHEET 2	13
5610-M-86		7	9.9-1	TURKEY POINT PLANT UNITS 3 & 4 CONTROL BUILDING HVAC EL 42'-0"	13
5610-M-87	1	4	9.9-2	TURKEY POINT PLANT UNITS 3 & 4 CONTROL BUILDING HVAC EL. 30'-0"	13
5610-M-301-12		32	7.7-3	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANEL"A" FRONT VIEW SECTION 3C04	13
5610-M-301-13	1	33	7.7-4	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANEL "A" FRONT VIEW SECTION 3C03	13

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FIGURE AND ENGR'G DRAWING CROSS-REFERENCES
(FOR DISTRIBUTION WITH FSAR UPDATE)

ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-M-301-20		28	7.7-9	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANELS "A" AND "C" FRONT VIEW SECTION 4C04	13
5610-M-301-23	1	24	7.7-7	TURKEY POINT PLANT UNITS 3 & 4 CONTROL CONSOLE FRONT VIEW SECTION 4C01	13
5610-M-301-23	2	9	7.7-8	TURKEY POINT PLANT UNITS 3 & 4 CONTROL CONSOLE FRONT VIEW SECTION 4C02	13
5610-M-301-26		25	7.7-10	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANEL "A" FRONT VIEW SECTION 4C03	13
5610-M-301-28	1	41	7.7-2a	TURKEY POINT PLANT UNITS 3 & 4 CONTROL CONSOLE EQUIPMENT LAYOUT SECTIONS 3C01	13
5610-M-301-28	2	9	7.7-2b	TURKEY POINT PLANT UNITS 3 & 4 CONTROL CONSOLE FRONT VIEW SECTION 3C02	13
5610-M-301-36		18	7.7-5	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANELS "B" AND "C" FRONT VIEW SECTION 3C05	13
5610-M-301-37		32	7.7-6	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANEL "B" FRONT VIEW SECTION 3C06	14
5610-M-301-40		19	7.7-11	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANEL "B" FRONT VIEW SECTION 4C05	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-M-301-41		34	7.7-12	TURKEY POINT PLANT UNITS 3 & 4 VERTICAL PANEL "B" FRONT VIEW SECTION 4C06	14
5614-M-724		0	1.2-8	TURKEY POINT PLANT UNIT 4 GENERAL ARRANGEMENT UNIT 4 EDG BUILDING PLAN AND SECTIONS	13
5610-M-1388		2	7.8-1	TURKEY POINT PLANT UNITS 3 & 4 LOOSE PARTS MONITORING SYSTEM	13
5610-M-3000	2	3	6.6-2	TURKEY POINT PLANT UNITS 3 & 4 LEGEND & GENERAL NOTES	13
5613-M-3008	1	10	9.6-8	TURKEY POINT PLANT UNIT 3 TURBINE PLANT COOLING WATER SYSTEM	13
5614-M-3008	1	14	9.6-9	TURKEY POINT PLANT UNIT 4 TURBINE PLANT COOLING WATER SYSTEM	13
5613-M-3010	1	11	10.2-60	TURKEY POINT PLANT UNIT 3 CIRCULATING WATER SYSTEM	13
5613-M-3010	2	6	10.2-61	TURKEY POINT PLANT UNIT 3 CIRCULATING WATER SYSTEM CONDENSER WATER BOX PRIMING	13
5613-M-3010	3	12	10.2-62	TURKEY POINT PLANT UNIT 3 CIRCULATING WATER SYSTEM LUBE WATER TO CIRCULATING WATER PUMPS	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-M-3010	1	11	10.2-63	TURKEY POINT PLANT UNIT 4 CIRCULATING WATER SYSTEM	13
5614-M-3010	2	6	10.2-64	TURKEY POINT PLANT UNIT 4 CIRCULATING WATER SYSTEM CONDENSER WATER BOX PRIMING	13
5614-M-3010	3	9	10.2-65	TURKEY POINT PLANT UNIT 4 CIRCULATING WATER SYSTEM LUBE WATER TO CIRCULATING WATER PUMPS	13
5613-M-3013	1	12	9.17-1	TURKEY POINT PLANT UNIT 3 INSTRUMENT AIR SYSTEM	14
5614-M-3013	1	9	9.17-2	TURKEY POINT PLANT UNIT 4 INSTRUMENT AIR SYSTEM	13
5613-M-3014	3	11	10.2-58	TURKEY POINT PLANT UNIT 3 CONDENSER SYSTEM	14
5614-M-3014	3	13	10.2-59	TURKEY POINT PLANT UNIT 4 CONDENSER SYSTEM	14
5610-M-3016	1	9	9.6A-5A	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION SYSTEM WATER SUPPLY AND STORAGE TANKS FLOW DIAGRAM	13
5610-M-3016	2	12	9.6A-5B	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION SYSTEM BACKUP SERVICE WATER FLOW DIAGRAM	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-M-3016	3	10	9.6A-5C	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION SYSTEM ELECTRIC AND DIESEL FIRE PUMPS FLOW DIAGRAM	14
5610-M-3016	4	3	9.6A-1	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION SYSTEM UNDERGROUND FIRE MAINS SITE LAYOUT PLAN	13
5610-M-3016	5	6	9.6A-2	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION SYSTEM TURBINE BLDG SPRINKLER SYSTEM BLDG LAYOUT PLAN	13
5610-M-3016	7	4	9.6A-3	TURKEY POINT UNITS 3 & 4 AUX. BLDG. AND EDG UNIT 3 DELUGE WATER SUPPRESSION DETAILS	13
5610-M-3016	9	0	9.6A-4	TURKEY POINT UNITS 3 & 4 FIRE PROTECTION SYSTEM HALON SUPPRESSION SYSTEM	13
5613-M-3018	1	13	9.11-11	TURKEY POINT PLANT UNIT 3 CONDENSATE STORAGE SYSTEM	14
5614-M-3018	1	15	9.11-10	TURKEY POINT PLANT UNIT 4 CONDENSATE STORAGE SYSTEM	14
5613-M-3019	1	18	9.6-1	TURKEY POINT PLANT UNIT 3 INTAKE COOLING WATER SYSTEM	13
5613-M-3019	2	15	9.6-2	TURKEY POINT PLANT UNIT 3 INTAKE COOLING WATER SYSTEM	13

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FIGURE AND ENGR'G DRAWING CROSS-REFERENCES
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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3019 AND FSAR FIGURE DELETED	3	N/A	9.6-3 DELETED	TURKEY POINT PLANT UNIT 3 INTAKE COOLING WATER SYSTEM TUBE CLEANING FOR CCW HEAT EXCHANGERS	13
5614-M-3019	1	22	9.6-5	TURKEY POINT PLANT UNIT 4 INTAKE COOLING WATER SYSTEM	14
5614-M-3019	2	14	9.6-6	TURKEY POINT PLANT UNIT 4 INTAKE COOLING WATER SYSTEM	13
5614-M-3019 AND FSAR FIGURE DELETED	3	N/A	9.6-7 DELETED	TURKEY POINT PLANT UNIT 4 INTAKE COOLING WATER SYSTEM TUBE CLEANING FOR CCW HEAT EXCHANGERS	13
5613-M-3020	1	11	9.6-15	TURKEY POINT PLANT UNIT 3 PRIMARY WATER MAKEUP SYSTEM	13
5613-M-3020	2	14	9.6-16	TURKEY POINT PLANT UNIT 3 PRIMARY MAKEUP WATER SYSTEM	13
5614-M-3020	1	11	9.6-17	TURKEY POINT PLANT UNIT 4 PRIMARY WATER MAKEUP SYSTEM	13
5614-M-3020	2	14	9.6-18	TURKEY POINT PLANT UNIT 4 PRIMARY MAKEUP WATER SYSTEM	13
5610-M-3021	1	10	9.6-10	TURKEY POINT PLANT UNITS 3 & 4 WATER TREATMENT PLANT SYSTEM FILTRATION	13

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5610-M-3021	2	9	9.6-11	TURKEY POINT PLANT UNITS 3 & 4 WATER TREATMENT PLANT SYSTEM DEMINERALIZER	13
5610-M-3021	3	6	9.6-12	TURKEY POINT PLANT UNITS 3 & 4 WATER TREATMENT PLANT SYSTEM DEMINERALIZER	13
5610-M-3021	4	5	9.6-13	TURKEY POINT PLANT UNITS 3 & 4 WATER TREATMENT PLANT SYSTEM WASTE NEUTRALIZATION	13
5610-M-3021	5	1	9.6-14	TURKEY POINT PLANTS UNITS 3 & 4 WATER TREATMENT PLANT SAMPLING SYSTEM	13
5613-M-3022	1	8	9.15-1	TURKEY POINT PLANT UNIT 3 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 3A AIR STARTING SYSTEM	13
5613-M-3022	2	7	9.15-2	TURKEY POINT PLANT UNIT 3 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 3B AIR STARTING SYSTEM	13
5613-M-3022	3	11	9.15-3	TURKEY POINT PLANT UNIT 3 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 3A FUEL OIL	13
5613-M-3022	4	7	9.15-4	TURKEY POINT PLANT UNIT 3 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 3B FUEL OIL	13
5613-M-3022	5	4	9.15-5	TURKEY POINT PLANT PLANT UNIT 3 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 3A LO & COOLING WATER	13

FSAR UPDATE REVISION 14
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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR REVISION NUMBER
5613-M-3022	6	4	9.15-6	TURKEY POINT PLANT UNIT 3 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 3B LO & COOLING WATER	13
5614-M-3022	1	2	9.15-7	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL ENGINE AND OIL SYSTEM EDG 4A AIR STARTING SYSTEM	14
5614-M-3022	2	2	9.15-8	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL ENGINE AND OIL SYSTEM EDG 4B AIR STARTING SYSTEM	14
5614-M-3022	3	1	9.15-9	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL ENGINE AND OIL SYSTEM EDG 4A FUEL SYSTEM	13
5614-M-3022	4	2	9.15-10	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL ENGINE AND OIL SYSTEM EDG 4B FUEL SYSTEM	13
5614-M-3022	5	3	9.15-11	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 4A LO & COOLING WATER	13
5614-M-3022	6	3	9.15-12	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL ENGINE AND OIL SYSTEM DG 4B LO & COOLING WATER	13
5610-M-3025	1	3	9.9-4	TURKEY POINT PLANT UNITS 3 & 4 CONTROL BUILDING VENTILATION CONTROL ROOM HVAC	13
5610-M-3025	2	8	9.9-5	TURKEY POINT PLANT UNITS 3 & 4 CONTROL BUILDING VENTILATION COMPUTER FACILITY/CABLE SPREADING ROOM HVAC	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG * REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3030	1	13	9.3-1	TURKEY POINT PLANT UNIT 3 COMPONENT COOLING WATER SYSTEM	14
5613-M-3030	2	8	9.3-2	TURKEY POINT PLANT UNIT 3 COMPONENT COOLING WATER SYSTEM	13
5613-M-3030	3	10	9.3-3	TURKEY POINT PLANT UNIT 3 COMPONENT COOLING WATER SYSTEM	13
5613-M-3030	4	17	9.3-4	TURKEY POINT PLANT UNIT 3 COMPONENT COOLING WATER SYSTEM	14
5613-M-3030	5	11	9.3-5	TURKEY POINT PLANT UNIT 3 COMPONENT COOLING WATER SYSTEM	14
5614-M-3030	1	16	9.3-6	TURKEY POINT PLANT UNIT 4 COMPONENT COOLING WATER SYSTEM	14
5614-M-3030	2	7	9.3-7	TURKEY POINT PLANT UNIT 4 COMPONENT COOLING WATER SYSTEM	13
5614-M-3030	3	15	9.3-8	TURKEY POINT PLANT UNIT 4 COMPONENT COOLING WATER SYSTEM	14
5614-M-3030	4	13	9.3-9	TURKEY POINT PLANT UNIT 4 COMPONENT COOLING WATER SYSTEM	14

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FIGURE AND ENGR'G DRAWING CROSS-REFERENCES
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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3033	1	11	9.3-10	TURKEY POINT PLANT UNIT 3 SPENT FUEL POOL COOLING SYSTEM	14
5614-M-3033	1	11	9.3-11	TURKEY POINT PLANT UNIT 4 SPENT FUEL POOL COOLING SYSTEM	14
5613-M-3034	1	3	9.8-3	TURKEY POINT PLANT UNIT 3 SPENT FUEL POOL AND NEW FUEL STORAGE AREA VENTILATION	13
5614-M-3034	1	2	9.8-4	TURKEY POINT PLANT UNIT 4 SPENT FUEL POOL AND NEW FUEL STORAGE AREA VENTILATION	14
5613-M-3036	1	13	9.4-1	TURKEY POINT PLANT UNIT 3 NUCLEAR STEAM SUPPLY SYSTEM SAMPLE SYSTEM	13
5614-M-3036	1	13	9.4-2	TURKEY POINT PLANT UNIT 4 NUCLEAR STEAM SUPPLY SYSTEM SAMPLE SYSTEM	13
5613-M-3041	1	16	4.2-1	TURKEY POINT PLANT UNIT 3 REACTOR COOLANT SYSTEM	13
5613-M-3041	2	20	4.2-9	TURKEY POINT PLANT UNIT 3 REACTOR COOLANT SYSTEM	13
5613-M-3041	3	19	4.2-10	TURKEY POINT PLANT UNIT 3 REACTOR COOLANT SYSTEM REACTOR COOLANT PUMPS	14

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FIGURE AND ENGR'G DRAWING CROSS-REFERENCES
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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3041	4	5	4.2-11	TURKEY POINT PLANT UNIT 3 REACTOR COOLANT SYSTEM PORV CONTROL	13
5614-M-3041	1	12	4.2-12	TURKEY POINT PLANT UNIT 4 REACTOR COOLANT SYSTEM	13
5614-M-3041	2	19	4.2-13	TURKEY POINT PLANT UNIT 4 REACTOR COOLANT SYSTEM	13
5614-M-3041	3	19	4.2-14	TURKEY POINT PLANT UNIT 4 REACTOR COOLANT SYSTEM REACTOR COOLANT PUMPS	14
5614-M-3041	4	5	4.2-15	TURKEY POINT PLANT UNIT 4 REACTOR COOLANT SYSTEM PORV CONTROL	13
5610-M-3046	1	19	9.2-1	TURKEY POINT PLANT UNITS 3 & 4 CHEMICAL AND VOLUME CONTROL SYSTEM BORIC ACID SYSTEM	14
5610-M-3046	2	16	9.2-2	TURKEY POINT PLANT UNITS 3 & 4 CHEMICAL & VOLUME CONTROL SYSTEM BORON RECYCLE SYSTEM	13
5610-M-3046	3	10	9.2-3	TURKEY POINT PLANT UNITS 3 & 4 CHEMICAL AND VOLUME CONTROL SYSTEM BORON RECYCLE SYSTEM	13
5610-M-3046	4	10	9.2-4	TURKEY POINT PLANT UNITS 3 & 4 CHEMICAL AND VOLUME CONTROL SYSTEM BORON RECYCLE SYSTEM	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3047	1	13	9.2-5	TURKEY POINT PLANT UNIT 3 CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN	13
5613-M-3047	2	21	9.2-6	TURKEY POINT PLANT UNIT 3 CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN	13
5613-M-3047	3	12	9.2-7	TURKEY POINT PLANT UNIT 3 CHEMICAL AND VOLUME CONTROL SYSTEM SEAL WATER INJECTION TO RCP	13
5614-M-3047	1	13	9.2-8	TURKEY POINT PLANT UNIT 4 CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN	13
5614-M-3047	2	24	9.2-9	TURKEY POINT PLANT UNIT 4 CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN	13
5614-M-3047	3	13	9.2-10	TURKEY POINT PLANT UNIT 4 CHEMICAL AND VOLUME CONTROL SYSTEM SEAL WATER INJECTION TO RCP	13
5613-M-3050	1	16	6.2-1	TURKEY POINT PLANT UNIT 3 RESIDUAL HEAT REMOVAL SYSTEM	14
5614-M-3050	1	17	6.2-5	TURKEY POINT PLANT UNIT 4 RESIDUAL HEAT REMOVAL SYSTEM	14
5613-M-3053	1	14	9.8-5	TURKEY POINT PLANT UNIT 3 CONTAINMENT PURGE SYSTEM AND PENETRATION COOLING SYSTEM	14

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-M-3053	1	12	9.8-6	TURKEY POINT PLANT UNIT 4 CONTAINMENT PURGE SYSTEM AND PENETRATION COOLING SYSTEM	14
5613-M-3057	1	5	9.10-1	TURKEY POINT UNIT 3 CONTAINMENT NORMAL AND EMERGENCY COOLING SYSTEMS	13
5614-M-3057	1	5	9.10-2	TURKEY POINT UNIT 4 CONTAINMENT NORMAL AND EMERGENCY COOLING SYSTEMS	13
5610-M-3060	1	8	9.8-1	TURKEY POINT PLANT UNITS 3 & 4 AUXILIARY BUILDING VENTILATION	13
5610-M-3060	2	2	9.8-2	TURKEY POINT PLANT UNITS 3 & 4 AUXILIARY BUILDING VENTILATION LAUNDRY DRYERS EXHAUST	13
5610-M-3061	1	12	11.1-9	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM WASTE HOLDUP & TRANSFER	14
5610-M-3061	2	7	11.1-10	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM LAUNDRY WASTE	13
5610-M-3061	3	5	11.1-11	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM DRAIN HEADERS AND SUMPS	13
5610-M-3061	4	4	11.1-12	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM POLISHING DEMINERALIZER	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-M-3061	5	9	11.1-13	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM WASTE EVAPORATOR FEED	13
5610-M-3061	6	6	11.1-14	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM WASTE EVAPORATOR PACKAGE	13
5610-M-3061	7	6	11.1-15	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM LIQUID SAMPLING, MONITORING, AND CHEMICAL ADDITION	13
5610-M-3061	8	4	11.1-16	TURKEY POINT PLANT UNITS 3 & 4 LIQUID WASTE DISPOSAL SYSTEM WASTE MONITOR TANKS	13
5610-M-3061	9	5	11.1-17	TURKEY POINT PLANT UNITS 3 & 4 SOLID WASTE DISPOSAL SYSTEM SPENT RESIN STORAGE	13
5610-M-3061	9 & 11	5 & 3	11.1-7	TURKEY POINT PLANT UNITS 3 & 4 RADWASTE SOLIDIFICATION SYSTEM CEMENT HANDLING AND CONTAINER FILLING	13
5610-M-3061	10	3	11.1-18	TURKEY POINT PLANT UNITS 3 & 4 SOLID WASTE DISPOSAL SYSTEM HOLDUP & MIXING	13
5610-M-3061	11	3	11.1-19	TURKEY POINT PLANT UNITS 3 & 4 SOLID WASTE DISPOSAL SYSTEM CONTAINER FILL	13
5610-M-3061	12	11	11.1-20	TURKEY POINT PLANT UNITS 3 & 4 GASEOUS WASTE DISPOSAL SYSTEM WASTE GAS COMPRESSORS	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5610-M-3061	13	5	11.1-21	TURKEY POINT PLANT UNITS 3 & 4 GASEOUS WASTE DISPOSAL SYSTEM WASTE GAS DECAY TANKS	13
5610-M-3061	14	6	11.1-22	TURKEY POINT PLANT UNITS 3 & 4 GASEOUS WASTE DISPOSAL SYSTEM GAS WASTE ANALYZERS	14
5613-M-3061	1	12	11.1-1	TURKEY POINT PLANT UNIT 3 LIQUID WASTE DISPOSAL SYSTEM REACTOR COOLANT DRAIN TANK AND PUMPS	13
5613-M-3061	2	3	11.1-2	TURKEY POINT PLANT UNIT 3 LIQUID WASTE DISPOSAL SYSTEM CONTAINMENT DRAINS	13
5614-M-3061	1	11	11.1-4	TURKEY POINT PLANT UNIT 4 LIQUID WASTE DISPOSAL SYSTEM REACTOR COOLANT DRAIN TANK AND PUMPS	13
5614-M-3061	2	3	11.1-8	TURKEY POINT PLANT UNIT 4 LIQUID WASTE DISPOSAL SYSTEM CONTAINMENT DRAINS	13
5613-M-3062	1	11	6.2-6	TURKEY POINT PLANT UNIT 3 SAFETY INJECTION SYSTEM	13
5613-M-3062	2	10	6.2-7	TURKEY POINT PLANT UNIT 3 SAFETY INJECTION SYSTEM	13
5614-M-3062	1	13	6.2-8	TURKEY POINT PLANT UNIT 4 SAFETY INJECTION SYSTEM	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-M-3062	2	11	6.2-9	TURKEY POINT PLANT UNIT 4 SAFETY INJECTION SYSTEM	13
5613-M-3064	1	14	6.2-10	TURKEY POINT PLANT UNIT 3 SAFETY INJECTION ACCUMULATOR SYSTEM INSIDE CONTAINMENT	13
5614-M-3064	1	17	6.2-11	TURKEY POINT PLANT UNIT 4 SAFETY INJECTION ACCUMULATOR SYSTEM INSIDE CONTAINMENT	13
5610-M-3065	2	7	10.2-57	TURKEY POINT PLANT UNIT 3 & 4 NITROGEN & HYDROGEN SYSTEMS NITROGEN CAP SYSTEM	13
5610-M-3065	3	7	9.2-11	TURKEY POINT PLANT UNITS 3 & 4 NITROGEN & HYDROGEN SYSTEMS HYDROGEN & CO2 SUPPLY	14
5613-M-3068	1	12	6.4-2	TURKEY POINT PLANT UNIT 3 CONTAINMENT SPRAY SYSTEM	13
5614-M-3068	1	10	6.4-3	TURKEY POINT PLANT UNIT 4 CONTAINMENT SPRAY SYSTEM	13
5613-M-3070	1	1	9.16-1	TURKEY POINT PLANT UNIT 3 TURBINE BUILDING VENTILATION LOAD CENTER & SWGR ROOMS CHILLED WATER SYSTEM-TRAIN A	13
5613-M-3070	2	1	9.16-2	TURKEY POINT PLANT UNIT 3 TURBINE BUILDING VENTILATION LOAD CENTER & SWGR ROOMS CHILLED WATER SYSTEM-TRAIN B	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-M-3070	1	1	9.16-3	TURKEY POINT PLANT UNIT 4 TURBINE BUILDING VENTILATION LOAD CENTER & SWGR ROOMS CHILLED WATER SYSTEM-TRAIN A	13
5614-M-3070	2	1	9.16-4	TURKEY POINT PLANT UNIT 4 TURBINE BUILDING VENTILATION LOAD CENTER & SWGR ROOMS CHILLED WATER SYSTEM-TRAIN B	13
5613-M-3072	1	23	10.2-1	TURKEY POINT PLANT UNIT 3 MAIN STEAM SYSTEM	13
5613-M-3072	2	8	10.2-2	TURKEY POINT PLANT UNIT 3 MAIN STEAM SYSTEM	13
5613-M-3072	3	8	10.2-3	TURKEY POINT PLANT UNIT 3 MAIN STEAM SYSTEM MSIV CONTROL	13
5614-M-3072	1	23	10.2-4	TURKEY POINT PLANT UNIT 4 MAIN STEAM SYSTEM	13
5614-M-3072	2	9	10.2-5	TURKEY POINT PLANT UNIT 4 MAIN STEAM SYSTEM	13
5614-M-3072	3	7	10.2-6	TURKEY POINT PLANT UNIT 4 MAIN STEAM SYSTEM MSIV CONTROL	13
5613-M-3073	1	13	10.2-15	TURKEY POINT PLANT UNIT 3 CONDENSATE SYSTEM	13

ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3073	2	11	10.2-16	TURKEY POINT PLANT UNIT 3 CONDENSATE SYSTEM	13
5613-M-3073	3	12	10.2-17	TURKEY POINT PLANT UNIT 3 CONDENSATE SYSTEM	13
5614-M-3073	1	14	10.2-18	TURKEY POINT PLANT UNIT 4 CONDENSATE SYSTEM	13
5614-M-3073	2	11	10.2-19	TURKEY POINT PLANT UNIT 4 CONDENSATE SYSTEM	13
5614-M-3073	3	15	10.2-20	TURKEY POINT PLANT UNIT 4 CONDENSATE SYSTEM	13
5610-M-3074	1	4	10.2-21	TURKEY POINT PLANT UNITS 3 & 4 FEEDWATER SYSTEM STANDBY STEAM GENERATOR FEEDWATER PUMPS	13
5610-M-3074	2	12	10.2-22	TURKEY POINT PLANT UNITS 3 & 4 FEEDWATER SYSTEM DEMINERALIZED STORAGE AND DEAERATION	14
5613-M-3074	1	9	10.2-23	TURKEY POINT PLANT UNIT 3 FEEDWATER SYSTEM	13
5613-M-3074	2	18	10.2-24	TURKEY POINT PLANT UNIT 3 FEEDWATER SYSTEM	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3074	3	12	10.2-25	TURKEY POINT PLANT UNIT 3 FEEDWATER SYSTEM	13
5613-M-3074	4	16	10.2-41	TURKEY POINT PLANT UNIT 3 FEEDWATER SYSTEM STEAM GENERATOR BLOWDOWN RECOVERY	14
5614-M-3074	1	11	10.2-26	TURKEY POINT PLANT UNIT 4 FEEDWATER SYSTEM	13
5614-M-3074	2	21	10.2-27	TURKEY POINT PLANT UNIT 4 FEEDWATER SYSTEM	13
5614-M-3074	3	13	10.2-28	TURKEY POINT PLANT UNIT 4 FEEDWATER SYSTEM	13
5614-M-3074	4	17	10.2-42	TURKEY POINT PLANT UNIT 4 FEEDWATER SYSTEM STEAM GENERATOR BLOWDOWN RECOVERY	14
5610-M-3075	1	12	9.11-2	TURKEY POINT PLANT UNITS 3 & 4 AUXILIARY FEEDWATER SYSTEM TURBINE DRIVE FOR AFW PUMPS	13
5610-M-3075	2	6	9.11-3	TURKEY POINT PLANT UNITS 3 & 4 AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER PUMPS	13
5613-M-3075	1	9	9.11-4	TURKEY POINT PLANT UNIT 3 AUXILIARY FEEDWATER SYSTEM STEAM TO AUXILIARY FEEDWATER PUMP TURBINES	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3075	2	8	9.11-5	TURKEY POINT PLANT UNIT 3 AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER TO STEAM GENERATORS	13
5613-M-3075	3	2	9.11-6	TURKEY POINT PLANT UNIT 3 AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES	13
5614-M-3075	1	7	9.11-7	TURKEY POINT PLANT UNIT 4 AUXILIARY FEEDWATER SYSTEM STEAM TO AUXILIARY FEEDWATER PUMP TURBINES	13
5614-M-3075	2	7	9.11-8	TURKEY POINT PLANT UNIT 4 AUXILIARY FEEDWATER SYSTEM AUXILIARY FEEDWATER TO STEAM GENERATORS	13
5614-M-3075	3	1	9.11-9	TURKEY POINT PLANT UNIT 4 AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES	13
5613-M-3077	1	9	10.2-47	TURKEY POINT PLANT UNIT 3 CONDENSATE POLISHING SYSTEM DEMINERALIZER	13
5613-M-3077	2	8	10.2-48	TURKEY POINT PLANT UNIT 3 CONDENSATE POLISHING SYSTEM DEMINERALIZER	13
5613-M-3077	3	7	10.2-49	TURKEY POINT PLANT UNIT 3 CONDENSATE POLISHING SYSTEM SPENT RESIN HANDLING SUBSYSTEM	13
5613-M-3077	4	2	10.2-50	TURKEY POINT PLANT UNIT 3 CONDENSATE POLISHING SYSTEM EFFLUENT SAMPLING	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-M-3077	1	7	10.2-51	TURKEY POINT PLANT UNIT 4 CONDENSATE POLISHING SYSTEM DEMINERALIZER	13
5614-M-3077	2	7	10.2-52	TURKEY POINT PLANT UNIT 4 CONDENSATE POLISHING SYSTEM DEMINERALIZER	13
5614-M-3077	3	5	10.2-53	TURKEY POINT PLANT UNIT 4 CONDENSATE POLISHING SYSTEM SPENT RESIN HANDLING SUBSYSTEM	13
5614-M-3077	4	3	10.2-54	TURKEY POINT PLANT UNIT 4 CONDENSATE POLISHING SYSTEM EFFLUENT SAMPLING	13
5613-M-3078	1	3	10.2-55	TURKEY POINT PLANT UNIT 3 STEAM GENERATOR WET LAYUP SYSTEM	13
5614-M-3078	1	4	10.2-56	TURKEY POINT PLANT UNIT 4 STEAM GENERATOR WET LAYUP SYSTEM	13
5613-M-3081	1	12	10.2-29	TURKEY POINT PLANT UNIT 3 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5613-M-3081	2	3	10.2-30	TURKEY POINT PLANT UNIT 3 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5613-M-3081	3	14	10.2-31	TURKEY POINT PLANT UNIT 3 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5613-M-3081	4	12	10.2-32	TURKEY POINT PLANT UNIT 3 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5613-M-3081	5	6	10.2-33	TURKEY POINT PLANT UNIT 3 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5613-M-3081	6	8	10.2-34	TURKEY POINT PLANT UNIT 3 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5614-M-3081	1	18	10.2-35	TURKEY POINT PLANT UNIT 4 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	14
5614-M-3081	2	6	10.2-36	TURKEY POINT PLANT UNIT 4 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5614-M-3081	3	13	10.2-37	TURKEY POINT PLANT UNIT 4 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5614-M-3081	4	13	10.2-38	TURKEY POINT PLANT UNIT 4 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5614-M-3081	5	6	10.2-39	TURKEY POINT PLANT UNIT 4 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13
5614-M-3081	6	6	10.2-40	TURKEY POINT PLANT UNIT 4 FEEDWATER HEATER SYSTEM FEEDWATER HEATER VENTS & DRAINS	13

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5613-M-3082	1	1	10.2-43	TURKEY POINT PLANT UNIT 3 SECONDARY SYSTEM WET LAYUP SYSTEM LOOP 1	13
5613-M-3082	2	5	10.2-44	TURKEY POINT PLANT UNIT 3 SECONDARY SYSTEM WET LAYUP SYSTEM LOOP 2	13
5614-M-3082	1	3	10.2-45	TURKEY POINT PLANT UNIT 4 SECONDARY SYSTEM WET LAYUP SYSTEM LOOP 1	13
5614-M-3082	2	5	10.2-46	TURKEY POINT PLANT UNIT 4 SECONDARY SYSTEM WET LAYUP SYSTEM LOOP 2	13
5613-M-3087	1	12	10.2-11	TURKEY POINT PLANT UNIT 3 TURBINE LUBE OIL SYSTEM LUBE & CONTROL OIL RESERVOIR	14
5613-M-3087	2	6	10.2-12	TURKEY POINT PLANT UNIT 3 TURBINE LUBE OIL SYSTEM LUBE & CONTROL OIL CONDITIONER	13
5614-M-3087	1	11	10.2-13	TURKEY POINT PLANT UNIT 4 TURBINE LUBE OIL SYSTEM LUBE & CONTROL OIL RESERVOIR	13
5614-M-3087	2	7	10.2-14	TURKEY POINT PLANT UNIT 4 TURBINE LUBE OIL SYSTEM LUBE & CONTROL OIL CONDITIONER	13
5613-M-3089	1	14	10.2-7	TURKEY POINT PLANT UNIT 3 STEAM TURBINE SYSTEMS	13

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5613-M-3089	2	15	10.2-8	TURKEY POINT PLANT UNIT 3 STEAM TURBINE SYSTEMS	13
5614-M-3089	1	15	10.2-9	TURKEY POINT PLANT UNIT 4 STEAM TURBINE SYSTEMS	13
5614-M-3089	2	15	10.2-10	TURKEY POINT PLANT UNIT 4 STEAM TURBINE SYSTEMS	13
5613-M-3094	1	19	9.12-1	TURKEY POINT PLANT UNIT 3 POST-ACCIDENT CONTAINMENT VENT AND SAMPLING SYSTEM FLOW DIAGRAM	14
5614-M-3094	1	17	9.12-2	TURKEY POINT PLANT UNIT 4 POST-ACCIDENT CONTAINMENT VENT AND SAMPLING SYSTEM FLOW DIAGRAM	14
5610-T-D-12A	1	8	7.2-9a	TURKEY POINT PLANT UNITS 3 & 4 ROD CONTROL SYSTEM CONTROL SYSTEM DIAGRAM	13
5610-T-D-12B	1	10	7.2-9b	TURKEY POINT PLANT UNITS 3 & 4 TAVG CONTROL AND INSERTION LIMIT ALARMS CONTROL SYSTEM DIAGRAM	14
5610-T-D-14	1	18	7.3-1	TURKEY POINT PLANT UNITS 3 & 4 REACTOR CONTROL SYSTEM CONTROL SYSTEM DIAGRAM	13
5610-T-D-15	1	19	7.2-12	TURKEY POINT PLANT UNITS 3 & 4 PRESSURIZER LEVEL CONTROL & PROTECTION AND CHARGING PUMP CONTROL CONTROL SYSTEM DIAGRAM	13

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5610-T-D-16A	1	20	7.2-11a	TURKEY POINT PLANT UNITS 3 & 4 PRESSURIZER PRESSURE PROTECTION & OVERPRESSURE MITIGATION SYSTEM CONTROL SYSTEM DIAGRAM	14
5610-T-D-16B	1	10	7.2-11b	TURKEY POINT PLANT UNITS 3 & 4 PRESSURIZER PRESSURE CONTROL CONTROL SYSTEM DIAGRAM	13
5610-T-D-17	1	23	7.2-13	TURKEY POINT PLANT UNITS 3 & 4 STEAM GENERATOR LEVEL CONTROL & PROTECTION CONTROL SYSTEM DIAGRAM	14
5610-T-E-1591	1	51	8.2-2	TURKEY POINT PLANT UNITS 3 & 4 MAIN AC DISTRIBUTION SYSTEM ONE LINE DIAGRAM	14
5610-T-L1	1	7	7.2-10	TURKEY POINT PLANT UNITS 3 & 4 INDEX AND SYMBOLS FOR LOGIC DIAGRAMS	13
5610-T-L1	2	18	7.2-5	TURKEY POINT PLANT UNITS 3 & 4 REACTOR PROTECTION SYSTEM REACTOR TRIP SIGNALS AND BREAKERS LOGIC DIAGRAM	13
5613-T-L1	9A1	2	8.2-18a	TURKEY POINT PLANT UNIT 3 EDG START SIGNALS LOGIC DIAGRAM	13
5614-T-L1	9A1	1	8.2-18L	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL GENERATOR START LOGIC DIAGRAM	13
5613-T-L1	9A2	2	8.2-18b	TURKEY POINT PLANT UNIT 3 EDG ENGINE START LOGIC DIAGRAM	13

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ENGINEERING DRAWING CROSS-REFERENCE	ENGR'G DRWG SHEET	DRWG REVISION IN FSAR	FSAR FIGURE NUMBER	FIGURE TITLE BLOCK	FSAR * REVISION NUMBER
5614-T-L1	9A2	1	8.2-18m	TURKEY POINT PLANT UNIT 4 EMERGENCY DIESEL GENERATOR ENGINE START LOGIC DIAGRAM	13
5613-T-L1	9A3	0	8.2-18c	TURKEY POINT PLANT UNIT 3 EDG VOLTAGE REGULATOR AND ELECTRIC GOVERNOR LOGIC DIAGRAM	13
5614-T-L1	9A3	1	8.2-18n	TURKEY POINT PLANT UNIT 4 DIESEL GENERATOR GOVERNOR & VOLTAGE REGULATOR CONTROL LOGIC DIAGRAM	13
5613-T-L1	9A4	2	8.2-18d	TURKEY POINT PLANT UNIT 3 EDG STOP/ENGINE SHUTDOWN LOGIC DIAGRAM	13
5614-T-L1	9A4	3	8.2-18o	TURKEY POINT PLANT UNIT 4 DIESEL ENGINE & GENERATOR STOP AND LOCKOUT LOGIC DIAGRAM	13
5613-T-L1	9A5	0	8.2-18e	TURKEY POINT PLANT UNIT 3 EDG LOCKOUT AND ENGINE AUXILIARIES LOGIC DIAGRAM.	13
5614-T-L1	9A5	2	8.2-18p	TURKEY POINT PLANT UNIT 4 DIESEL GENERATOR BREAKER AND FUEL OIL PUMP CONTROL LOGIC DIAGRAM	13
5613-T-L1	9A6	0	8.2-18f	TURKEY POINT PLANT UNIT 3 EDG BREAKER CONTROL LOGIC DIAGRAM	13
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5613-T-L1	9A7	0	8.2-18g	TURKEY POINT PLANT UNIT 3 EDG INDICATIONS AND ALARMS LOGIC DIAGRAM	13
5614-T-L1	9A7	1	8.2-18r	TURKEY POINT PLANT UNIT 4 DIESEL ENGINE & GENERATOR ANNUNCIATION AND INDICATION LOGIC DIAGRAM - SHEET 2	13
5610-T-L1	16	10	7.4-2a	TURKEY POINT PLANT UNITS 3 & 4 NUCLEAR INSTRUMENTATION TRIP SIGNALS LOGIC DIAGRAM	13
5610-T-L1	17	13	7.4-2b	TURKEY POINT PLANT UNITS 3 & 4 NUCLEAR INSTRUMENTATION PERMISSIVES AND BLOCKS LOGIC DIAGRAM	13
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NOTE *: THIS IS THE FSAR REVISION THAT INCORPORATED THE CROSS-REFERENCED ENGINEERING DRAWING REVISION.

Superseded Pages per Rev. 14 to FSAR. Letter dated 5/12/97
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Revised 7/19/97 (gh)

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1.1 SITE AND ENVIRONMENT

The site is on the shore of Biscayne Bay, about 25 miles south of Miami, Florida. The area immediately surrounding the site is low and swampy and is very sparsely populated, with much of it unsuited for development without raising the elevation with fill. The nearest farming area lies in the northwest quarter of a 5-mile arc from the site.

The area surrounding the site is flat and slopes very gently to the west from sea level at the shoreline of Biscayne Bay to an elevation of about 10 ft above MSL at a point some 8 to 10 miles inland. To the east across Biscayne Bay from 5 to 8 miles, is a series of offshore islands running in a northeast-southwest direction between the Bay and the Atlantic Ocean, the largest of which is Elliott Key.

The site is well ventilated with air movement prevailing almost 100 percent of the time. The atmosphere in the area is generally unstable with diurnal inversions of short duration.

The Miami area has experienced winds of hurricane force periodically. During storms the plant may be subjected to flood tides of varying heights. Hurricane "Betsy" in 1965 produced the maximum flooding recorded, which was about 10 feet above MSL. External flood protection is described in Appendix 5G.

The normal direction of natural drainage of surface and ground water in the area of the site is to the east and south toward Biscayne Bay and will not affect off-site wells. A radiological background study of the Turkey Point

area will be initiated approximately one year prior to initial startup of the Unit 3. This will involve the collection of samples of air, soil, water, marine life, biota and vegetation in the area. The bed rock beneath the limerock fill is competent with respect to foundation conditions for the nuclear units. The area is in a seismologically quiet region, all of Florida being classified Zone 0 (the zone of least probability of damage) by the Uniform Building Code, as published by International Conference of Building Officials.

11.3 GENERAL DESIGN CRITERIA

The general design criteria define or describe safety objectives and approaches incorporated in.the design. These general design criteria are addressed explicitly in the pertinent sections in this report. The remainder of this section, 1.3, presents a brief description of related features which are provided to meet the design objectives reflected in the criteria. The description is developed more fully in those succeeding sections of the report indicated by the references.

The parenthetical numbers following the section headings indicate the numbers of the 1967 proposed draft General Design Criteria (GDC).

1.3.1 OVERALL REQUIREMENTS (GDC 1-GDC 5)

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of an accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Quality standards of material selection, design, fabrication and inspection conform to the applicable provisions of recognized codes, and good nuclear practice.

All systems and components designated Class I are designed so that there is no loss of capability to perform their safety function in the event of the maximum hypothetical seismic ground acceleration acting in the horizontal and vertical directions simultaneously. The working stress for Class I item is kept within code allowable values for the design seismic ground acceleration. Similarly, measures are taken in the design to protect against high winds,

sudden barometric pressure changes, flooding, and other natural phenomena. The Containment and Auxiliary Building are designed to withstand the effects of a tornado.

Reference sections:

Section Title	Section
Site and Environment; Meteorology, Seismology	2.7, 2.9
Reactor Coolant System; Design Bases	4.1
Containment Structure; Design Bases	5.1
Electrical System; Design Bases	8.1
Unit 4 Emergency Diesel Generator Building	5.3.4
Structures, Systems and Equipment	Appendix 5A

The fire protection program for the nuclear units is described in the below referenced section:

Reference section:

Section Title	Section
Fire Protection Program	Appendix 9.6A

Certain components of the Auxiliary, Emergency and Waste Disposal Systems are shared by Units 3 and 4. Certain components of shared equipment may be called upon to fulfill either an emergency, or emergency and shutdown function. The design and its evaluation supports the capability to deal with the affected unit, while maintaining safe control of the second unit.

A complete set of as-built drawings is maintained throughout the life of the units. A set of all the quality assurance data generated during fabrication and erection of the essential components is retained.

Reference section:

Section Title	Section
Records	12.4
Initial Tests and Operation	13
Functional Evaluation of the Components of the Systems which are shared by the two units	Appendix A

1.3.2 PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS (GDC 6-GDC 10)

The reactor core with its related control and protection system is designed to function throughout its design lifetime without exceeding acceptable fuel limits specified to preclude damage. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

The Reactor Control and Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than 1.30.

Reference sections:

Section Title	Section
Reactor, Design Basis	3.1, 3.2
Instrumentation and Control, Protective Systems	7.2
Safety Analysis	14

The design of the reactor core and related protection systems ensures that power oscillations which could cause fuel damage in excess of acceptable limits are not possible.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It was concluded that low frequency xenon oscillations may occur in the axial dimension and part length control rods were provided to suppress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor xenon induced oscillations. The part length control rods were removed from the core after the first few cycles of operation. Their removal was based on a determination that their presence was not required, since the control banks provide adequate means for controlling the xenon oscillations.

The moderator temperature and overall power coefficient in the power operating range is maintained negative by inclusion of burnable poison in the first core loading.

Reference section:

Section Title	Section
Reactor Design, Nuclear Design and Evaluation	3.2.1
Reactor Coolant System Pipe Rupture	14.3

The Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

The materials of construction of the pressure boundary of the Reactor Coolant System are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical.

When the reactor is critical, means for showing the relative reactivity status of the reactor is provided by control bank positions displayed in the control room. Periodic samples of the coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

Instrumentation and controls provided for the protective systems are designed to trip the reactor, when necessary, to prevent or limit fission product release from the core and to limit energy release; to signal containment isolation; and to control the operation of engineered safety features equipment.

During reactor operation in the startup and power modes, redundant safety limit signals will automatically actuate two reactor trip breakers which are in series with the rod drive mechanism coils. This action would interrupt power and initiate reactor trip.

Reference section:

Section Title	Section
Instrumentation and Controls	7.1, 7.2, 7.4, 7.7

If the reactor protection system receives signals which are indicative of an approach to an unsafe operating condition, the system actuates alarms, prevents control rod motion, initiates load cutback, and/or opens the reactor trip breakers.

The basic reactor operating philosophy is to define an allowable region of power and coolant temperature conditions. This allowable range is defined by the primary tripping functions, the overpower ΔT trip, over-temperature ΔT trip, and the nuclear overpower trip. The operating region below these trip settings is designed so that no combination of power, temperatures and pressure could result in DNBR less than 1.30 with all reactor coolant pumps in operation. Additional tripping functions such as a high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water level trip,
loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low level trip, turbine trip, safety injection trip, nuclear source and intermediate range level trips, and manual trip are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

Rod stops from nuclear overpower, overpower ΔT and over-temperature ΔT deviation are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the reactor control system or by operator error.

Reference sections:

Section Title	Section
Engineered Safety Features	6.2
Reactor Protection System	7.2

Positive indications in the control room of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of the containment air activity. Deviations from normal containment environmental conditions including air particulate activity, radiogas activity, and, in the case of gross leakage, the liquid inventory in the process systems and containment sump, will be detected.

The Reactor Protection System is capable of protecting against any single anticipated malfunction of the reactivity control system and is designed to limit reactivity transients to DNBR 1.30 due to any single malfunction in the deboration controls.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates of the order of 12×10^{-4} Δk/sec which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

Reference sections:

Section Title	Section
Reactor Design Bases	3.1
Protection Systems	7.2
Regulating Systems	7.3
Chemical and Volume Control System	9.2

1.3.6 REACTOR COOLANT PRESSURE BOUNDARY (GDC 33-GDC 36)

The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection.

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The following general criteria are followed to assure conservatism in computing the required containment structural load capacity:

a) In calculating the containment pressure, rupture sizes up to and including a double-ended break of reactor coolant pipe are considered.

b) In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated including failures of a diesel-generator, an emergency containment cooler and a containment spray pump.

c) The pressure and temperature loadings obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and design wind or seismic forces, do not exceed the load-carrying capacity of the structures, its access openings or penetrations.

The reinforced concrete containment is not susceptible to a low temperature brittle fracture. The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between $50^{o}F$ and $120^{o}F$. Operation with elevated normal bulk containment temperatures up to $125^{o}F$ for short periods of time during the summer months has been evaluated (See Section 14.0). The material for the containment penetrations, which are designed to Subsection B of Section III ASME B&PV Code has a NDT of $0^{o}F$.

The reactor coolant pressure boundary does not extend outside of the containment. Isolation valves for all fluid system lines penetrating the containment provide at least two barriers against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation, and maintenance of the systems and to ensure reliable operation of other engineered safety features.

After completion of the containment structure an initial integrated leak rate test is conducted at the calculated peak accident pressure, to verify that the leakage rate is not greater than 0.25 per cent by weight of the containment volume per day.

Leak rate tests, using the same method as the initial leak rate test, will be performed during unit shutdowns periodically in accordance with the Technical Specifications.

1.5 DESIGN HIGHLIGHTS

The design of Turkey Point Units 3 and 4 is based upon proven concepts which have been developed and successfully applied in the construction of pressurized water reactor system. In subsequent paragraphs, a few of the design features are listed which represent slight variation or extrapolations from units presently operating such as San Onofre and Connecticut-Yankee.

1.5.1 POWER LEVEL

The license application power level of 2200 MWt is larger than the capability of the Connecticut Yankee plant and is a reasonable increase over power levels of pressurized water reactors now operating.

1.5.2 REACTOR COOLANT LOOPS

The Reactor Coolant System for the Turkey Point Units 3 and 4 consists of three loops as compared with four loops for Connecticut-Yankee. The use of three loops for the production of 2200 MWt requires an attendant increase in the size and capacity of the Reactor Coolant System components such as the reactor coolant pumps, piping and steam generators. These increases represent reasonable engineering extrapolations of existing proven designs.

1.5.3 PEAK SPECIFIC POWER

The design rating is slightly higher than that licensed in CVTR (17 kw/ft) and slightly lower than that of Saxton (19.1 kw/ft). The maximum overpower condition is 20.0 kw/ft (112%) compared to 20 kw/ft (118%) for CVTR.

1.5.4 FUEL ASSEMBLY DESIGN

The fuel assembly design incorporates the rod cluster control concept in a canless assembly utilizing a spring clip grid to provide support for the 15 x 15 array of fuel rods. This concept incorporates the advantages of the Yankee canless fuel assembly and the Saxton spring clip with the rod cluster control scheme. Extensive out-of-pile tests have been performed on this concept and operating experience is available from the San Onofre and Connecticut-Yankee plants.

1.5.5 ENGINEERED SAFETY FEATURES

The engineered safety features provided are of the same types provided for the Connecticut-Yankee plant augmented by borated water injection accumulators. A Safety Injection System is provided which can be operated from emergency on-site diesel power. An Emergency Cooling and Filtering System is provided for post-loss-of-coolant conditions. A Containment Spray System provides cool, borated water spray into the containment atmosphere for additional cooling capacity.

1.5.6 EMERGENCY POWER

In addition to the multiple ties to offsite power sources, four emergency diesel generators are provided as emergency power supplies for the case of loss of offsite power. The emergency diesel generators are capable of operating sufficient safety injection and containment cooling equipment to ensure an acceptable post-loss-of-coolant pressure transient for any credible single failure.

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TABLE 3.2.2-1

THERMAL AND HYDRAULIC DESIGN PARAMETERS

Total Primary Heat Output, MWt	2208
Total Reactor Coolant Pump Heat Output, MWt	8
Total Core Heat Output, MWt	2200
Total Core Heat Output, Btu/hr	7508.6×10^6
Heat Generated in Fuel, %	97.4
Maximum Thermal Overpower, %	12
Nominal System Pressure, psia	2250
Hot Channel Factors (First cycle)*	
Heat Flux	
Nuclear, F^N_q	3.13
Engineering, F^E_q	1.03
Total, F_q	3.23
Enthalpy Rise	
Nuclear, $F^N_{\Delta H}$	1.75
Engineering, $F^E_{\Delta H}$	1.01
Total, $F_{\Delta H}$	1.77
Coolant Flow	
Total Flow Rate, lb/hr	101.5×10^6
Average Velocity Along Fuel Rods, ft/sec	14.3 (14.0)**
Average Mass Velocity, lb/hr-ft^2	2.32×10^6 (2.28×10^6)**
Coolant Temperature, ^oF	
Nominal Inlet	546.2
Average Rise in Vessel	55.9
Average Rise in Core	58.3 (59.3)**
Average in Core	575.4 (575.9)**
Average in Vessel	574.2
Nominal Outlet of Hot Channel	642.0 (643.2)**
Heat Transfer (First Cycle)	
Active Heat Transfer Surface Area, ft^2	42,460
Average Heat Flux, Btu/hr-ft^2	171,600
Maximum Heat Flux, Btu/hr-ft^2	554,200
Maximum Thermal Output, kw/ft	17.9
Maximum Clad Surface Temperature at Nominal Pressure, ^oF	657
Maximum Average Clad Temperature at Rated Power, ^oF	715
Fuel Central Temperatures, ^oF (First Cycle)	
Maximum at 100% Power	4150
Maximum at 112% Power	4400
DNB Ratio (First Cycle)	
Minimum DNB Ratio at nominal operating conditions	1.81
Pressure Drop, psi (First Cycle)	
Across Core	26
Across Vessel, including nozzles	46

* - See note in Table 3.2.1-1 for subsequent cycles.
** - Values for complete thimble plug removal.

TABLE 3.2.2-2

ENGINEERING HOT CHANNEL FACTORS

(FIRST CYCLE)

F_q^E	Pellet Diameter, Density Enrichment, and Eccentricity Rod Diameter, (Pitch and Bowing)	1.03
$F_{\Delta H}^E$	Pellet Diameter, Density, Enrichment Rod Diameter, Pitch and Bowing	1.08
	Inlet Flow Maldistribution	1.01
	Flow Redistribution	1.03
	Flow Mixing	0.90*
	Resulting $F_{\Delta H}^E$	1.01

*To point of Minimum DNB ratio

3.2.3 MECHANICAL DESIGN AND EVALUATION

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2.3-1 and in elevation in Figure 3.2.3-2. The core, consisting of the fuel assemblies, control rods, source rods and burnable poison rods provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. A listing of the core mechanical design parameters is given in Table 3.2.3-1.

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The LOPAR and OFA assemblies are nearly identical in their geometric configuration (number of fuel rods and thimble tubes, fuel rod dimensions, assembly pitch, etc.) with the following exceptions: 1) the diameter of the upper portion of the OFA guide thimble tubes has been reduced, relative to the LOPAR assembly, to accommodate the increased Zircaloy-4 grid strap thickness and results in reduced diameter thimble plugs, and; 2) the five intermediate support grids are of different materials. The enrichment of each region of fuel will vary slightly depending on the energy requirements for a given cycle of operation.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of 144 inches within Zircaloy-4 tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The enrichments of the fuel for the various regions in the first core are given in Table 3.2.3-1. Enrichment for subsequent cycles are given in the cycle specific RSE. All fuel rods are internally pressurized with helium during fabrication. Heat generated by the fuel is removed by demineralized borated light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The stress criteria of Article 4 Section III of the ASME code is employed in the design of the fuel assembly with the exception of the fuel clad which is specifically excluded by the code. The criteria for the design of fuel rods

are listed in Section 3.1.3. Zircaloy-4 which is used for fabricating the guide thimbles of the fuel assembly and Inconel 718 which is used for fabricating grids and assembly hold down springs are not yet considered as code materials. In LOPAR fuel, all grids are made of Inconel-718. In OFA fuel, intermediate grids are made of Zircaloy-4 and the top and bottom grids of Inconel-718. The method for establishing design stress intensity values for the materials is consistent with that outlined in the code.

the corner legs. The ligaments between the holes of the nozzle plate are positioned laterally beneath the fuel rods to prevent passage of the rods beyond this surface.

The RCC guide thimble tubes, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle end plate. These loads as well as the weight of the assembly are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins.

The OFA bottom nozzle assembly is essentially the same design as the LOPAR bottom nozzle, except for the instrumentation tube counterbore diameter being reduced. This reduction accommodates a reduced outside diameter on the OFA guide thimble tube and provides the same radial clearance with the OFA instrumentation tube as the LOPAR assembly nozzle. This assures retention of the instrumentation tube lower end.

The reconstitutable OFA bottom nozzle design has a feature which allows it to be easily removed. As shown in Figure 3.2.3-9B, a locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire used for the standard LOPAR nozzle design. The reconstitutable nozzle design facilitates removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

Top Nozzle

The top nozzle is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is directed toward the flow holes in the upper core plate. The nozzle is comprised of an adapter plate, enclosure, top plate, two clamps, four leaf springs, and assorted hardware. All parts with the exception of the springs and their hold down bolts are constructed of Type 304 stainless steel. The springs are made from Inconel and the bolts are made of a nickel chromium alloy.

The adaptor plate is square in cross-section, and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the top ends of the control guide thimble tubes are fastened to the adaptor. Thus, the adaptor plate acts as the fuel assembly top end plate, and provides a means of distributing evenly among the guide thimble tubes any axial loads imposed on the fuel assemblies..

The OFA top nozzle is the same as the LOPAR assembly top nozzle with the exception of a small increase in the adaptor plate thickness. This increase results in a slightly longer OFA length of .055 inches as shown in Figure 3.2.3-9A. The increased adapter plate thickness is a result of a standardization of the OFA nozzle design. This minor change has no adverse effect on the OFA/LOPAR assemblies fuel handling operation or mixed-core operations.

The nozzle enclosure is actually a square thin walled tubular shell which forms the plenum section of the top nozzle. The bottom end of the enclosure is pinned and welded to the periphery of the adaptor plate, and the top end is welded to the periphery of the top plate.

The top plate is square in cross-section with a central hole. The hole allows clearance for the RCC absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to

unsupported span between the fuel assembly adaptor plate and the end of the guide tube in the upper internals package. The spiders which support the source rods and burnable poison rods are all contained within the fuel top nozzle. Beginning with the Turkey Point Unit 3 Cycle 14 reload (Region 16), a keyless/cuspless top nozzle and holddown spring change was implemented. The keyless/cuspless top nozzle is functionally interchangeable with the old design.

Guide Thimbles

The control rod guide thimbles in the fuel assembly provide guided channels for the absorber rods during insertion and withdrawal of the control rods. They are fabricated from a single piece of Zircaloy 4 tubing, which is drawn to two different diameters. The larger inside diameter at the top provides a relatively large annular area for rapid insertion during a reactor trip and to accommodate a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is of reduced diameter to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

Flow holes are provided just above the transition of the two diameters to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is closed at the bottom by means of a welded end plug. The end plug is fastened to the bottom nozzle during fuel assembly fabrication.

Grids

The grid assemblies consist of individual slotted straps which are assembled and interlocked in an "egg-crate" type arrangement and then furnace brazed to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes, and tabs are punched and formed in the individual straps prior to assembly.

Two types of grid assemblies are used in the fuel assembly. One type of these grids having mixing vanes which project from the edges of the straps into the coolant stream is used in the high heat region of the fuel assemblies to promote mixing of the coolant. A grid of this type is shown in Figure 3.2.3-10. The other type of grids, located at the bottom and top ends of the assembly, are of the nonmixing type. They are similar to the mixing type with the exception that they contain no mixing vanes on the internal straps.

There are two materials used to construct support grids for the LOPAR and OFA assemblies. Inconel-718 is used for all seven LOPAR grids, and the top and bottom non-mixing, support grids in the OFA assembly. Zircaloy is used for the five intermediate mixing-vane grids in the OFA assembly. A more detailed description can be found in the Reload Transition Safety Report (RTSR) for Turkey Point Units (Reference 2).

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose.

Inconel-718 and Zircaloy are for the grid material because of their corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material is age hardened to obtain the material strength necessary to develop the required grid spring forces.

Impact tests that have been performed at 600°F to obtain the dynamic strength data verify that the Zircaloy grid strength data at reactor operating conditions is structurally acceptable. The OFA Zircaloy grid design has approximately 7-percent less crush strength than the Inconel grid design, and both grid designs maintain their integrity during the most severe load conditions of a combined seismic/LOCA event.

Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in a slightly cold worked Zircaloy-4 tubing which is plugged and seal welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without over-stressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a stainless steel helical compression spring which bears on the top of the fuel.

The fuel rods employed on LOPAR and OFA assemblies are geometrically identical with only slight variations in some design parameters. The Debris Resistant Fuel Assembly (DRFA), a modified OFA with debris resistant features, utilizes a lower end plug which is approximately 1.4 inches longer than in the standard OFA design. On a cycle-to-cycle and region-to-region basis, fuel enrichment, plenum void volume and initial helium backfill pressure will vary somewhat to accommodate specific cycle design requirements. This fact was also applicable prior to the introduction of OFA assemblies. For Unit 3 beginning with cycle-12, the DRFA incorporates axial blankets which consist of low enriched or natural uranium oxide pellets extending 6 inches at the top and bottom of the fuel stack within the fuel rod. Unit 4 has axial blankets starting with Cycle 14 and DRFA starting with Cycle 13.

During fuel rod assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process. A hold-down force in excess of the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack.

The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide powder which is compacted by cold pressing and sintering to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length. The ends of each OFA fuel pellet have a small chamfer at the outer cylinder surface.

The pellet densities are adjusted as shown in Table 3.2.3-1 to compensate for the effects of the higher burnup of fuel in regions remaining longest in the core. A different fuel enrichment as listed in Table 3.2.3-1 is used for each of the three regions in the first core loading. Reload region, as-built fuel enrichments and pellet densities are provided in each applicable Reload Safety Evaluation (RSE) Report.

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, meticulous process control is exercised.

Process Control

Powder withdrawal from storage can be made by one authorized group only who direct the powder to correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single enrichment and density are produced in a given production line.

Finished pellets are placed on trays having the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Physical barriers prevent mixing of pellets of different densities and enrichments in this storage area. Unused powder and substandard pellets are returned to storage in the original color coded containers.

Each fuel assembly will be identifed by means of a serial number engraved on the upper nozzle. The fuel pellets will be fabricated by a batch process so that only one enrichment region is processed at any given time. The serial numbers of the assemblies and corresponding enrichment will be documented by the manufacturer and verified prior to shipment.

Each assembly will be assigned a core loading position. A record will then be made of the core loading position, serial number and enrichment. Prior to core loading, independent checks will be made to ensure that this assignment is correct.

During initial core loading and subsequent refueling operations, detailed handling and checkoff procedures will be utilized throughout the sequence. The initial core will be loaded in accordance with the core loading diagram similar to Figure 3.2.3-3 which shows the location for each of the three enrichment types of fuel assemblies in the region. Reload cycle core loading diagrams are provided in the cycle specific RSE Report.

Rod Cluster Control Assemblies

The rod cluster control assemblies (RCCA) each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies one of which is shown in Figure 3.2.3-4 are provided to control the reactivity of the core under operating conditions. These assemblies consist of rods containing full length absorber material. The number of RCC assemblies is specified in Table 3.2.3-1.

The absorber material used in the control rods is silver-indium-cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCC assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents, and detents for connection to the drive shaft, are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the RCC assembly and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub, and the fingers are joined to the vanes by furnace brazing. A centerpost which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in the service. All components of the spider assembly are made from Type 304 stainless steel except for the springs which are Inconel X-750 alloy and the retainer which is of 17-4 Ph material.

The absorber rods are secured to the spider so as to assure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating of assembly misalignments.

In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage ensuring their effectiveness under all operating conditions.

Neutron Source Assemblies

Four neutron source assemblies were utilized in the initial core. They consisted of two secondary source assemblies each, and two primary source assemblies each. The rods in the source assembly are fastened to a spider at the top end similar to the RCC spiders.

In the core, the neutron source assemblies are inserted into the RCC guide thimbles in fuel assemblies at unrodded locations. The location of these assemblies in the core is shown in Figure 3.2.3-3. The number and location of secondary source assemblies is given in each cycle specific RSE report.

The primary and secondary source rods both utilized the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing, 0.432 in O.D. with 0.019 inch thick walls). The secondary source rods contain Sb-Be pellets stacked to a height of 121.75 inches. Design criteria similar to those for the fuel rods are used for the design of the source rods; ie, the cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and cladding.

Thimble Plug Assemblies

Evaluations have been performed to support the complete or partial removal of thimble plugs from Turkey Point Units 3 & 4. These evaluations have addressed the effect of thimble plug removal on Core Design, Core Thermal Hydraulics, Reactor Pressure Vessel System thermal hydraulics and the non-LOCA and LOCA safety analyses. Based on these evaluations, it has been determined that it is acceptable to remove all or any combination of thimble plugs from Turkey Point Units 3 & 4.

The thimble plug assemblies as shown in Figure 3.2.3-10A consist of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Similar short rods are also used on the source assemblies and burnable poison assemblies to plug the ends of all vacant fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adaptor plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a small lock-pin welded to the nut.

The OFA thimble plug has a smaller diameter (0.485 inch) than the LOPAR thimble plug diameter (0.498 inch) in order to maintain the same thimble plug to thimble tube diametral clearance, and to limit bypass flow through the OFA guide thimbles while providing sufficient coolant flow to cool the core components.

All components in the thimble plug assembly, except for the springs, are constructed from type 304 stainless steel. The springs are wound from an age hardened nickel base alloy for corrosion resistance and high strength.

Burnable Poison Rod

The burnable poison rods are statically suspended and positioned in vacant RCC thimble tubes within the fuel assemblies at nonrodded core locations. The poison rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat spider plate which fits within the fuel assembly top nozzle and rests on the top adaptor plate.

The spider plate and the poison rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the poison rods cannot be lifted out of the core by flow forces.

Several types of burnable absorbers are currently utilized in Turkey Point Units 3 and 4. Typically, LOPAR assemblies are both full-length and part-length borosilicate burnable poison rods which consist of borosilicate glass tubes contained within type 304 stainless steel cladding which is plugged and sealed at both ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall type 304 stainless steel, tubular inner liner (Figure 3.2.3-11).

The second major type is the Wet Annular Burnable Absorber[3] (WABA) (Figure 3.2.3-11A) which will, as necessary, be used with OFA assemblies. The WABA consists of an annular aluminum oxide-boron carbide (Al_2O_3-B_4C) absorber clad in two concentric Zircaloy tubes. Coolant flows through the center holes as well as through the outer annulus between the WABA and the guide thimble tube. The WABA design provides significantly enhanced nuclear characteristics compared with the borosilicate absorber rod design. Fuel cycle benefits result from the reduced parasitic nature of Zircaloy compared to stainless steel tubes, increased water fraction in the burnable absorber cell, and a reduced penalty at the end of each cycle.

The third major type of neutron absorber being used is the Hafnium Vessel Flux Depression (HVFD) absorber (Figure 3.2.3-14). The HVFD consists of a reduced length annular hafnium absorber axially positioned within Zircaloy cladding. The primary function of the HVFD is to provide for reactor vessel flux reduction to satisfy pressurized thermal shock considerations[1].

The fourth major type of burnable absorber currently utilized is the Integral Fuel Burnable Absorber (IFBA). The IFBA rods have a thin (1.77 mg/in) boride coating on the cylindrical surface of the fuel pellets along the central portion of the fuel stack length. In order to offset the effects of the Helium gas release from the IFBA coating during irradiation, a lower initial Helium backfill pressure is used in the IFBA rods compared to the non-IFBA fuel rods.

Visual examination of these rods during 1982 refueling shutdown revealed satisfactory mechanical integrity.

Four demonstration rods are being irradiated for a second cycle in the Indian Point 3 reactor in order to demonstrate integrity during extended duty, and will be re-examined following completion of their second cycle during 1985.

The rods are designed in accordance with the standard fuel rod design criteria; i.e., the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass as a result of the B_{10} (n,a) reaction. The large void volume required for the helium is obtained through the use of glass in tubular form which provides a central void along the length of the rods. A more detailed discussion of the borosilicate glass BP rod design is found in WCAP 9000 (4).

Based on available data on properties of Pyrex glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should possibly occur. The top end of the inner liner is open to receive the helium which diffuses out of the glass.

To ensure the integrity of the burnable poison rods, the tubular cladding and end plugs are procured to the same specifications and standard of quality as is used for stainless steel fuel rod cladding and end plugs in other Westinghouse plants. In addition, the end plug seal welds are checked for integrity by visual inspection and x-ray. The finished rods are helium leak checked.

Removable Rod Assemblies

Four demonstration assemblies were loaded in Turkey Point Unit 3, Cycle 10. Each of the assemblies contains twenty-eight demonstration Integral Fuel Burnable Absorber (IFBA) fuel rods and one hundred and seventy-six unpoisoned fuel rods.

The mechanical design of the demonstration assemblies is identical to that of the other fuel assemblies in the reload region, except that the demonstration assemblies are tne "removable rod" type, which will allow removal of some rods for post-irradiation inspections. Similar "removable rod" type Optimized Fuel Assemblies have been used in previous demonstration assembly programs at Farley, Salem, Beaver Valley and Point Beach reactors. The mechanical design of the assemblies has been evaluated and meets the same acceptance criteria as the standard fuel assembly design for steady state, transient, seismic and LOCA conditions.

The design of the fuel rods contained in the demonstration assemblies is identical to the fuel rod design of the other fuel rods in the reload region except that:

1. in eacn of the demonstration assemblies, there are fifty-two removable fuel rods (sixteen removable IFBA rods, 36 removable non-IFBA rods); these removable fuel rods have longer, more slender top end plugs to facilitate rod removal and a larger chamfer on their bottom end plugs to ease fuel rod reinsertion;

2. for the IFBA fuel rods only (sixteen removable IFBA fuel rods and twelve non-removable IFBA fuel rods per assembly), each fuel stack contains absorber material coated on the outside diameter of the UO_2 fuel pellets and distributed uniformly over the entire fuel stack; because the burnable absorber material releases additional helium into the fuel rod during depletion, the IFBA fuel rods are prepressurized to 200 psig during manufacture, whereas the non-IFBA fuel rods in the demonstration assemblies, and the standard fuel rods throughout the reload region, are prepressurized to 350 psig.

3. the core locations of the IFBA demonstration assemblies were chosen such that the IFBA fuel rods are never the lead power rods.

Based on review of the appropriate phase diagram and on destructive examination after one reactor cycle of test rods incorporating coated pellets essentially identical in material and manufacture method, no adverse chemical interaction of the absorber material with either cladding or fuel pellet is predicted for the times and temperatures of operation.

The approved fuel rod model (PAD) [5] was used to assess in detail the fuel rod design criteria influenced by addition of the absorber material. Based upon a consideration of clad stress, fuel temperatures, and rod internal pressure, an allowable burnup for the demonstration rods in excess of the planned burnup of fuel assemblies was calculated. No adverse effects on fuel rod performance are predicted.

Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods is calculated by the PAD code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to this phenomenon is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

Th maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than 1 per cent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, tensile and ultrasonic testing of fuel tubes, dimensional inspection, ultrasonic test or x-ray of both end plug welds, gamma scanning and helium leak tests.

In the event of cladding defects, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature or fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of fuel elements behavior in high burnup uranium dioxide[6] indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considers the effect of burnup, temperature distribution and internal voids. It is an empirical model which has been checked with data from numerous operating Westinghouse reactors.

The integrity of fuel rod cladding, is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate strength 57,000 psi minimum	Yield strength 45,000 psi minimum
Strain	1.7%	1.0%

The damage limits given above are minimum values. Actual damage limits depend upon neutron exposure and normal variation of material properties and would generally be greater than these minimum damage limits.

For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. Internal gas pressure:

   The internal gas pressure of the lead rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady-state operation, and which could cause extensive DNB propagation to occur. The safety evaluation of the fuel rod internal pressure design basis is presented in Reference 7.

2. Cladding temperature:

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature through the wall during normal operating conditions is given in Table 3.2.2-1.

3. Burnup:

   Fuel burnup results in fuel swelling which produces cladding strain. The strain damage limit is not expected to be reached until the peak pin burnup is in excess of 60,000 MWD/MTU. Peak pin burnup is calculated during the design process and maintained below 60,000 MWD/MTU.

4. Fuel temperature and kw/ft:

At zero burnup, cladding damage is calculated to occur at 31 kw/ft based upon cladding strain reaching the damage limit. At this power rating 17% of the pellet central region is expected to be in the molten condition. The maximum thermal output is much less as shown in Table 3.2.2-1.

The use of chamfered fuel pellets in Optimized Fuel Assemblies results in a hot spot average fuel temperature increase of less than 20°F compared to unchamfered pellets. Evaluation results show that all core design criteria and safety limits (including LOCA and non-LOCA transients) are satisfied when using chamfered pellets.

Evaluation of Burnable Poison Rods

The burnable poison rods are positioned in the core inside fuel assembly guide thimbles and held down in place by attachment to a plate assembly compressed beneath the upper core plate and hence cannot be the source of any reactivity transient. Due to the low heat generation rate, and the conservative design of the poison rods, there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients including loss of coolant.

Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCCA's, and burnable poison rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for an infinite number of cycles.

In the case of the fuel grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small ($\approx$ .001) and the stress associated with the motion is significantly small (< 100 psi). Likewise, the reactions at the grid spring due to the motion is much less than the preload spring force and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and the grid support are not anticipated.

The effect of thermal cycling on the grid-clad support is merely a slight relative movement between the grid contact surfaces and the clad, which is gradual in nature during heat-up and cool-down. Since the number of cycles of the occurrence is small over the life of a fuel assembly ($\approx$ 6 years), negligible wear of the mating parts is expected.

In-core operation of assemblies in the Yankee Rowe and Saxton reactors using similar clad support have verified the calculated conclusions. Additional test results under simulated reactor environment in the Westinghouse Reactor Evaluation Channel also support these conclusions.

The dynamic deflection of the full length control rods and the burnable poison rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot. With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in either clad tubing or in the flexure joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider, or the retainer plate in the case of the burnable poison rods.

<u>References Section 3.2.3</u>

1. Letter from Uhrig, R. E., FP&L to Varga, S. A., NRC, Subject: Pressurized Thermal Shock, Letter No L-83-180, March 25, 1983

2. Petrarca, D., et al, "Reload Transition Safety Report For Turkey Point Units 3 & 4", June, 1983

3. Letter from Thomas, C. O., NRC, to Rahe, E. P., Westinghouse, Subject: Acceptance for Referencing of Licensing Topical Report WCAP-10021 (P), Revision 1, and WCAP-10377 (NP), "Westinghouse Wet Annular Burnable Absorber Evaluation Report", August 9, 1983

4. WCAP-9000 (1968), "Nuclear Design of Westinghouse Pressurized Water Reactor with Burnable Poison Rods", PROPRIETARY. A NON-PROPRIETARY version of this report is WCAP-7806, Rev. 1.

5. WCAP-8720, Addendum-2, "Westinghouse Revised Pad Code Thermal Safety Model", October 27, 1982 (Proprietary)

6. Daniel, R. C., et al. "Effects of High Burnup on Zircaloy-Clad Bulk DO_2 Plate Fuel Element Samples," WAPD-263, (September, 1965)

7. George, R. A., Lee, Y. C., Eng, G. H. "Revised Clad Flattening Model," WCAP-8377 (Westinghouse Proprietary) and WCAP-8381 (Non-Proprietary), July 1974.

8. XN-NF-85-12(P) Ford, K. L., et al "Mechanical Design Report For Turkey Point Units 3 & 4 Hafnium Vessel Flux Depression (HVFD) Cluster Assemblies", Proprietary.

9. WCAP-12346, "Turkey Point Units 3 and 4 - 15x15 Debris Resistant Fuel Assembly Design Report," July 1989.

TABLE 3.2.3-1

CORE MECHANICAL DESIGN PARAMETERS[1]

Active Portion of the Core

Equivalent Diameter, in.	119.7
Active Fuel Height, in - Unit 3	144.00, 144.00, 143.474
Unit 4	144.00, 143.40, 142.80
Length-to-Diameter Ration	1.2
Total Cross Section Area, $Ft.^2$	78.1

Fuel Assemblies

Number 157	
Rod Array 15 x 15	
Rods per Assembly	204 (2)
Rods Pitch, in.	0.563
Overall Dimensions, in.	8.426 x 8.426
Fuel Weight (as UO_2), pounds	176,000
Total Weight, pounds	225,000
Number of Grids per Assembly	7
Guide Thimble I.D. (Above Dashpot), in.	0.512
(at Dashpot), in.	0.455

Fuel Rods

Number	32,028
Outside Diameter, in.	0.422
Diametral Gap, mils	7.5, 7.5, 8.5
Clad Thickness, in.	0.0243
Clad Material	Zircaloy-4
Overall Length, in. Unit 3	152.060
Unit 4	152.360

Fuel Pellets

Material	UO_2 sintered
Density (% of Theoretical) - First Cycle (3)	
Region 1	94 (10.3 g/cc)
Region 2	93 (10.19 g/cc)
Region 3	92 (10.08 g/cc)(Unit 4-93)
Fuel Enrichments w/o - First Cycle (3)	
Region 1	1.85
Region 2	2.55
Region 3	3.10
Diameter, in. - Unit 3 (Regions 1, 2, 3)	0.3659, 0.3659, 0.3649
Unit 4 (All Regions)	0.3659
Length, in.	0.600

NOTES :

(1) All Dimensions are for cold conditions

(2) Twenty-one rods are omitted: twenty to provide passage for control rods and one to contain in-core instrumentation

(3) Values for current cycles are given in Appendixes 14A and 14B.

TABLE 3.2.3-1

<u>Rod Cluster Control Assemblies</u>	
Neutron Absorber	5% Cd, 15% In, 80% Ag
Cladding Material	Type 304 SS - Cold Worked
Clad Thickness, in.	0.019
Number of Clusters	45
Full Length	45
Number of Control Rods per Cluster	20
Weight in 60°F Water	
Full Length, pounds	147
Length of Rod Control, in.	158.454 (overall) 150.574 (insertion length)
Length of Absorber Section, in.	142.00
<u>Core Structure</u>	
Core Barrel, in.	
I.D.	133.875
O.D.	137.875
Thermal Shield, in.	
I.D.	142.625
O.D.	148.0
<u>Burnable Poision Rods</u> (4)	
Number	816
Material	Borosilicate Glass
Outside Diameter, in.	0.4395
Inner Tube, O.D. in.	0.2365
Clad Material	S.S.
Inner Tube Material	S.S.
Boron Loading (natural) gm/cm of glass rod	0.0429
<u>Neutron Source Assemblies</u> (5)	
Primary Source (typical)	Pu-Be
Secondary Source (typical)	Sb-Be

NOTES :

(4) Values for current cycles are given in Appendices 14A and 14B.

(5) The actual neutron source installed are described in the Reload Safety Evaluation for each specific cycle.

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TABLE 4.1-1

REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

Total Primary Heat Output, MWt	2208
Total Primary Reat Output, Btu/hr	7535×10^6
Number of Loops	3
Coolant Volume (liquid), including total pressurizer volume, ft^3	9343
Total Reactor Coolant Flow, gpm	265,500
	Pressure, psiq
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485 ±1%
Power Relief Valves	
i) Normal Operation	2335
ii) OMS Actuation During Heatup and Cooldown	
a) RCS ≤ 285°F	415 ±15
	Setpoint increases step-wise[1]:
b) RCS 319°F	495
RCS 347°F	600
RCS 384°F	832.5
Rcs 421°F	1147.5
RCS 472°F	1710
RCS 508°F	2220
RCS 554°F	2335
RCS 750°F	2335
Pressurizer Spray Valves (Open)	2260
High Pressure Trip	2385
High Pressure Alarm	2310
Low Pressure Trip	1835
Low Pressure Alarm	2185
Hydrostatic Test Pressure	3107

1. OMS is not normally in-service at RCS temperatures greater than 285°F.

TABLE 4.1-2

REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3107
Design Temperature, °F	650
Overall Height of Vessel and Closure Head, ft-in. (Bottom Head O.D. to top of Control Rod Mechanism Housing)	42-7
Water Volume, (with core and internals in place), ft^3	3667
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	58
Diameter of Reactor Closure Head Studs, in.	6
Flange, ID, in.	149.6
Flange, OD, in.	184
ID at Shell, in.	155.5
OD across inlet/outlet nozzles, in.	230-5/16 / 240
Inlet Nozzle ID, in.	Tapered 27-15/32 to 33-13/16
Outlet Nozzle ID, in.	28.97
Clad Thickness, min., in.	0.156
Lower Head Thickness, min., in.	4-3/4 plus cladding
Vessel Belt-Line Thickness, min., in.	7-3/4 plus cladding
Closure Head Thickness, in.	6-3/16 plus cladding
Reactor Coolant Inlet Temperature, °F	546.2
Reactor Coolant Outlet Temperature, °F	602.1
Reactor Coolant Flow, lb/hr	101.5×10^6

TABLE 4.1-8

DESIGN THERMAL AND LOADING CYCLES - 40 YEARS

Transient Design Condition Cycles	Design Cycles	Expected
1. Station heatup at 100°F per hour	200 (5/yr)	80
2. Station cooldown at 100°F per hour	200 (5/yr)	80
3. Station loading at 5% of full power/min	14,500 (1/day)	2500
4. Station unloading at 5% of full power/min	14,500 (1/day)	2500
5. Step load increase of 10% of full power (but not to exceed full power)	2000 (1/week)	500
6. Step load decrease of 10% of full power	2,000 (1/week)	500
7. Step load decrease of 50% of full power	200 (5/year)	20
8. Reactor trip	400 (10/year)	40
9. Hydrostatic test at 3107 psig pressure, 100°F temperature	5 (pre-operational)	2
10. Hydrostatic test at 2435 psig pressure and 400°F temperature	150 (post-operational)	30

11. Steady state fluctuations - the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur.

TABLE 4.1-9

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

Component	Codes
Reactor Vessel	ASME III* Class A
Control Rod Drive Mechanism Housings	ASME III* Class A
Steam Generator	
Tube Side	ASME III* Class A
Shell Side ***	ASME III* Class C
Reactor Coolant Pump Casing	No Code (Design per ASME III-Article 4)
Pressurizer	ASME III* Class A
Pressurizer Relief Tank	ASME III* Class C
Pressurizer Safety Valves	ASME III*
Reactor Coolant Piping	ASA B31.1**
System valves, fittings and piping	ASA B31.1**
Core Exit Thermocouple Seal Assemblies (Head Port Adapters & Drive Sleeves)	ASME III* Subsection NB, Class 1, 1986 Edition

* ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** ASA B31.1-1955 Code for Pressure Piping, plus Code Cases N-7 and N-10 where applicable.

*** The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ±10% and field results have shown the repeatability of the trip point to be within ±1%. The analysis of the loss of flow transient presented in Section 14.1.9 assumes instrumentation error of ±3%.

4.2.10 REACTOR COOLANT SUBCOOLED MARGIN MONITOR

The reactor coolant system subcooled margin monitor system is an on-line microcomputer based system which uses reactor coolant process signals to provide a continuous indication of the margin from saturation conditions. The subcooled margin monitor system also provides an alarm signal into the main control room annunciator.

The reactor coolant system parameters monitored are the three coolant loops hot leg temperature, and loops A and B hot leg pressure. The operator has the choice of continuous main control board indication of either the pressure or temperature margin from saturation.

The temperature sensors are dual RTD's installed in thermowells. These RTD's are connected to provide the subcooling margin monitor system computing module with a 4-20 ma dc signal.

The reactor coolant pressure transmitters also provide a 4-20 ma dc signal to the computing module.

The computing module selects the highest temperature from those provided and the lowest pressure and calculates the margin to saturation from those two readings. The readings then appear on the display module in the control room.

4.2.11 REACTOR COOLANT VENT SYSTEM.

The RCS vent system provides the operator with a means to vent non-condensible gases from the Reactor Coolant System. As shown on Figure 4.2-1 and 4.2-5, the RCS can be vented separately through the reactor vessel head vent or from the pressurizer steam space via the pressurizer relief line.

To vent system discharges to the containment sump and/or the pressurizer relief tank.

The RCS vent system can vent one-half of the RCS volume (gas) in one hour at operating pressure, but is sized such that the RCS mass inventory will be maintained by the charging pumps should the vent line suffer a guillotine break.

The power for the vent valves is taken from vital DC power outside the containment. Valve control and position indication is located in the control room. Pressure indication is provided in the control room to assist the operator in determining leakage in the vent line. Each vent is powered from an emergency bus.

The vent system has been seismically analyzed.

4.2.12 REACTOR VESSEL DRAIN LEVEL INDICATION SYSTEM

The reactor vessel drain down level indication system (see Figure 4.2-1) provides the continuous measurement of reactor coolant level during drain down operations and while in a drain down condition. This system also provides audible and visual alarm annunciation on increasing reactor vessel level above a preset volume. The system consists of two independent and redundant level (differential pressure) transmitters with control room indication. Audio and visual alarms are located in the control room and an audio alarm (horn) and light is located at each steam generator manway.

TABLE 4.3-1

SUMMARY OF PRIMARY PLUS SECONDARY STRESS INTENSITY
FOR COMPONENTS OF THE REACTOR VESSEL

Area	Stress Intensity (psi)	Allowable Stress 3 S_m (psi) (Operating Temperature)
Control Rod Housing	25,600	69,900
Head Flange	51,000	80,000
Vessel Flange	63,000	80,000
Closure Studs	82,300	110,250
Outlet Nozzles	45,000	80,000
Inlet Nozzles & Vessel Supports	38,000	80,000
Core Support pad	68,794	69,900
Bottom head to shell jucture	24,000	80,000
Bottom instrumentation	69,200	69,900
Vessel Wall Transition	26,000	80,000

TABLE 4.3-2

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR
COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*
Control rod housing	0.0
Head Flange	0.005
Vessel Flange	0.011
Stud bolts	0.231
Outlet nozzles	0.42638
Inlet nozzles & Vessel support	0.20957
Core support pad	0.00125
Bot. head to shell juncture	0.000
Bot. instrumentation	0.0
Vessel Wall Transition	0.0

*As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 4.3-2a

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR
PRESSURE BEARING COMPONENTS OF THE REACTOR COOLANT PUMPS

Item	Usage Factor
Casing	<0.001
Main Flange	0.025
Bolts	0.26

4.4 TESTS AND INSPECTIONS

4.4.1 REACTOR COOLANT SYSTEM INSPECTION

Non-Destructive Inspection of Materials and Components

Table 4.4-1 summarizes the quality assurance program for all Reactor Coolant System components. In this table all of the non-destructive tests and inspections which are required by Westinghouse specifications on Reactor Coolant System components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which are more stringent in some areas than those requirements specified in the applicable codes, are also included.

Westinghouse requires, as part of its reactor vessel specification, that certain special tests which are not specified by the applicable codes be performed. These tests are listed below:

1) Ultrasonic Testing - Westinghouse requires that a 100% volumetric ultrasonic test of reactor vessel plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates are required by code to receive only a longitudinal wave ultrasonic test on a 9 in. x 9 in. grid. The 100% volumetric ultrasonic test is a severe requirement, but it assures that the plate is of the highest quality.

2) Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and ypost-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) test specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and are in accordance with the version of ASTM E185,"Recommended Practice for Surveillance Tests on Structural Material in Nuclear Reactors," required by 10 CFR 50, Appendix H.

The reactor vessel surveillance programs used eight original specimen capsules, which are located about 3 inches from the vessel wall directly opposite the center portion of the core. Reference is made to Section 3.2.3.

The capsules are periodically removed and evaluated to determine changes in material properties. The surveillance specimen withdrawal schedule is shown in Table 4.4-2. The capsules contain reactor vessel steel specimens from the shell plates or forgings located in the core region of the reactor and associated weld metal and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM Committee E10 Radioisotopes and Radiation Effects are inserted in the capsules. The eight original capsules contained approximately 27 tensile specimens, 256 Charpy V-notch specimens (which will include weld metal and heat affected zone material) and 44 WOL specimens. Dosimeters including Ni, Cu, Fe, Co Al, Cd shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238 are placed in filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and insure good thermal conductivity. The complete capsule is helium leak tested.

Irradiation of the specimens will be higher than the irradiation of the vessel because the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and are actual samples from the materials used in the vessel, the NDTT measurements will be representative of the vessel at a later time in life. Data from fracture toughness samples (WOL) are expected to provide additional information for use in determining allowable stresses for irradiated material.

TABLE 4.4-2 [1]

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM
WITHDRAWAL SCHEDULE

TURKEY POINT UNIT 3

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRWAL TIME
U	30°	0.49	Standby
V	---	----	Specimen withdrawal at 12 Years
W	40°	0.34	Standby
X	270°	2.48	33 Years
Y	150°	0.49	Standby
Z	230°	0.34	Standby

TURKEY POINT UNIT 4

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRWAL TIME
U	30°	0.49	Standby
V	290°	0.79	24 Years
W	40°	0.34	Standby
X	270°	2.48	Standby
Y	150°	0.49	Standby
Z	230°	0.34	Standby

NOTES:

1. This table was originally Technical Specification Table 4.4-5, which was referenced in Surveillance Requirement 4.4.9.1.2.