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L-2011-028
10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Safety Analyses Issues – Round 1

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.

(2) Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU - Reactor Systems (SRXB) Requests for Additional Information – Round 1," Accession No. ML110460085, February 15, 2011

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

By email from the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM) dated February 15, 2011 [Reference 2], additional information regarding the Steam Generator Tube Rupture (SGTR) Margin-To-Overfill (MTO) analysis, Best Estimate Large Break Loss of Coolant Accident (LBLOCA) analysis, and Turkey Point's (PTN) General Design Criteria (GDC) 30 on Reactor Holddown Capability was requested by the NRC staff in the Reactor Systems Branch (SRXB) to support their review of the EPU License Amendment Request (LAR). The Request for Additional Information (RAI) consisted of seven (7) questions: five (5) questions regarding the SGTR MTO analysis, one (1) question regarding the Best Estimate LBLOCA analysis, and one (1) question regarding PTN GDC 30 requirements. These seven RAI questions and the applicable FPL responses are documented in the Attachment to this letter.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State Designee of Florida.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2009-133 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

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NRR

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 16, 2011.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health

Turkey Point Units 3 and 4

RESPONSE TO NRC RAI REGARDING EPU LAR NO. 205
AND SRXB SAFETY ANALYSES ISSUES – ROUND 1

ATTACHMENT 1

Response to Request for Additional Information

The following information is provided by Florida Power & Light (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter (L-2010-113) dated October 21, 2010 [Reference 1].

In an email dated February 15, 2011 [Reference 2], the NRC staff requested additional information regarding FPL's request to implement the Extended Power Uprate. The RAI consisted of seven (7) questions from the NRC's Reactor Systems Branch (SRXB): five (5) questions regarding the Steam Generator Tube Rupture (SGTR) Margin to Overfill (MTO) analysis, one (1) question regarding the Best Estimate Large Break Loss-of-Coolant Accident (LBLOCA) analysis, and one (1) question regarding PTN GDC-30 requirements on Reactor Holddown Capability. These seven RAI questions are documented below with the applicable FPL responses.

Steam Generator Tube Rupture

SRXB-1.1: Provide a thermal hydraulic analysis for Turkey Point at the proposed, uprated conditions, for a limiting margin-to-overfill/overfill scenario. One acceptable methodology would be for the analysis to align as closely as possible to what is approved in WCAP-10698-P-A; however, since the licensee has asserted that a limiting single failure is not in the Turkey Point licensing basis, this exception to the WCAP-10698-P-A methodology would be acceptable. Consider limiting single failures and discuss what they could be.

FPL has performed analyses of the limiting margin-to-overfill scenario for operation at the proposed Extended Power Uprate (EPU) core power level of 2644 MWt. The analysis aligned closely to WCAP-10698-P-A. Exceptions are discussed in response SRXB 1.2 below. However it is recognized that a single failure assumption is not in the Turkey Point licensing basis, therefore it is an exception from the consideration of limiting single failures discussed in WCAP-10698-P-A methodology.

The analyses were performed using the LOFTTR2 thermal hydraulic model consistent with the methodology in WCAP-10698-P-A.

In addition to the changes made to incorporate the modeling presented in WCAP-10698-P-A, updated operator action times to remove excess conservatism in the MTO analysis have also been implemented. These operator response times during recovery from a SGTR event were recorded using the plant training simulator with various operating crews. The times were tabulated and a bounding set of response times were selected for use in the margin to overfill analysis. Table 2 shows the comparison between the operator action times used on Reference 1 and the times used in the revised analysis. These simulator-based action times have been modeled in the LOFTTR2 analysis to predict the dynamic system response to the Turkey Point specific recovery actions.

FPL has a plant simulator and training programs which provide the required assurance that the necessary actions and times can be taken consistent with those assumed for the WCAP-10698-P-A design basis analysis.

The results indicate a margin to overfill greater than 300 ft3 in the ruptured steam generator (SG) for the EPU case. No water is transferred into the steam lines. The sequence of events for the revised analysis is provided in Table 1. Figures 1, 2, and 3 provide the time-dependent primary and secondary pressures, primary-to-secondary break flow, and ruptured steam generator water volume, respectively, for the limiting EPU scenario.

Table 1: Limiting MTO Scenario Sequence of Events

Event	EPU Time (sec)
Tube Rupture	0
Reactor Trip and LOOP	102
AFW Initiation	103
SI Actuation	113
Ruptured SG AFW Isolation	403
Reduce SI Pumps Running	704
Isolate Ruptured SG MSIV	1304
Initiate Cooldown with Intact SG	1784
Establish Charging Flow	1788
Terminate Cooldown	2060
Initiate Depressurization	2420
Terminate Depressurization	2476
Terminate SI Flow	2656
Balance Charging and Letdown Flows	2780
Break Flow Termination	3132

Table 2: SGTR Operator Action Times

Action	EPU Time (as provided in Reference 1)	Revised EPU Time
Operator action time to isolate auxiliary feedwater flow to the ruptured steam generator following reactor trip	5 minutes	5 minutes
Operator action time to isolate safety injection flow from two of the four safety injection pumps following reactor trip	18 minutes	10 minutes
Operator action time to close main steam isolation valve to isolate steam flow from the ruptured steam generator following reactor trip	27 minutes	20 minutes*
Operator action time to initiate cooldown	10 minutes (following isolation of the ruptured steam generator)	28 minutes (after reactor trip)
Operator action time to establish maximum charging flow	Start of cooldown OR 37 minutes from reactor trip**	Start of cooldown OR 28 minutes from reactor trip**
Plant response to complete cooldown	LOFTTR2-calculated	LOFTTR2-calculated
Operator action time to initiate depressurization following completion of cooldown	5 minutes	6 minutes
Plant response to complete depressurization	LOFTTR2-calculated	LOFTTR2-calculated
Operator action time to terminate ECCS flow following completion of depressurization	3 minutes	3 minutes
Operator action time to balance letdown and charging flow following safety injection termination	2 minutes	2 minutes
Plant response until break flow termination resulting from primary and secondary pressure equalization	LOFTTR2-calculated	LOFTTR2-calculated

* Required to be closed prior to initiation of cooldown. Not an explicit operator response time.
** The assumption of a minimum time to perform this step decreases the margin to overfill the SG and results in a conservative assumption in the analysis.

Figure 1: RCS and Secondary Pressures (EPU)

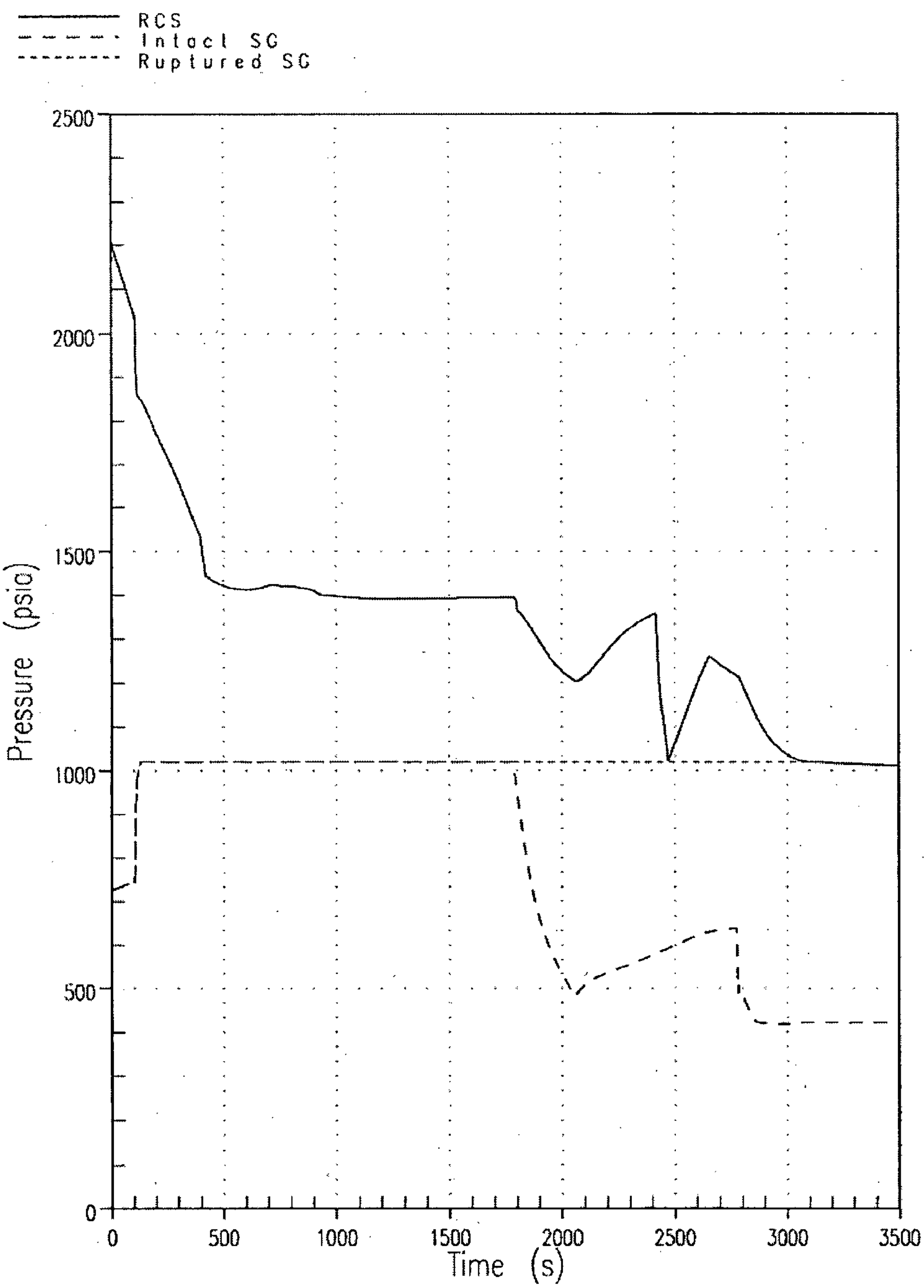


Figure 2: Ruptured Steam Generator Break Flow (EPU)

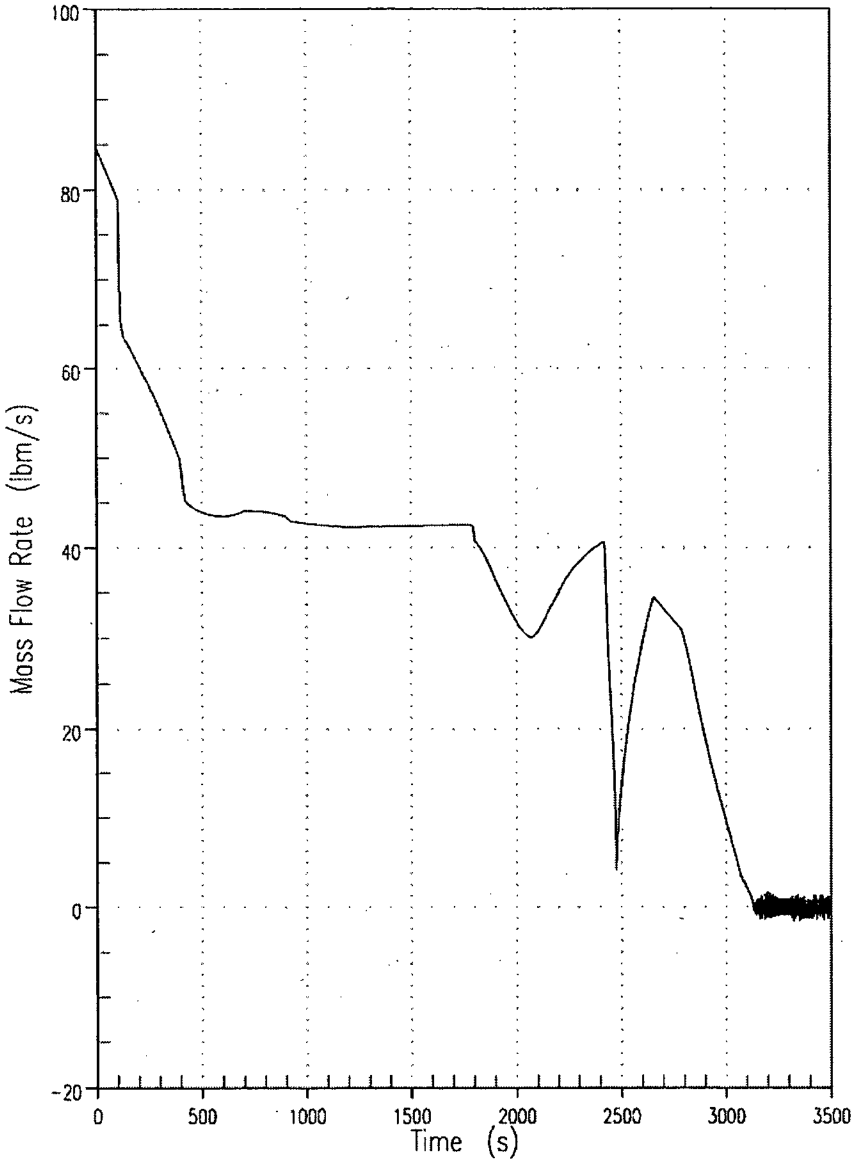
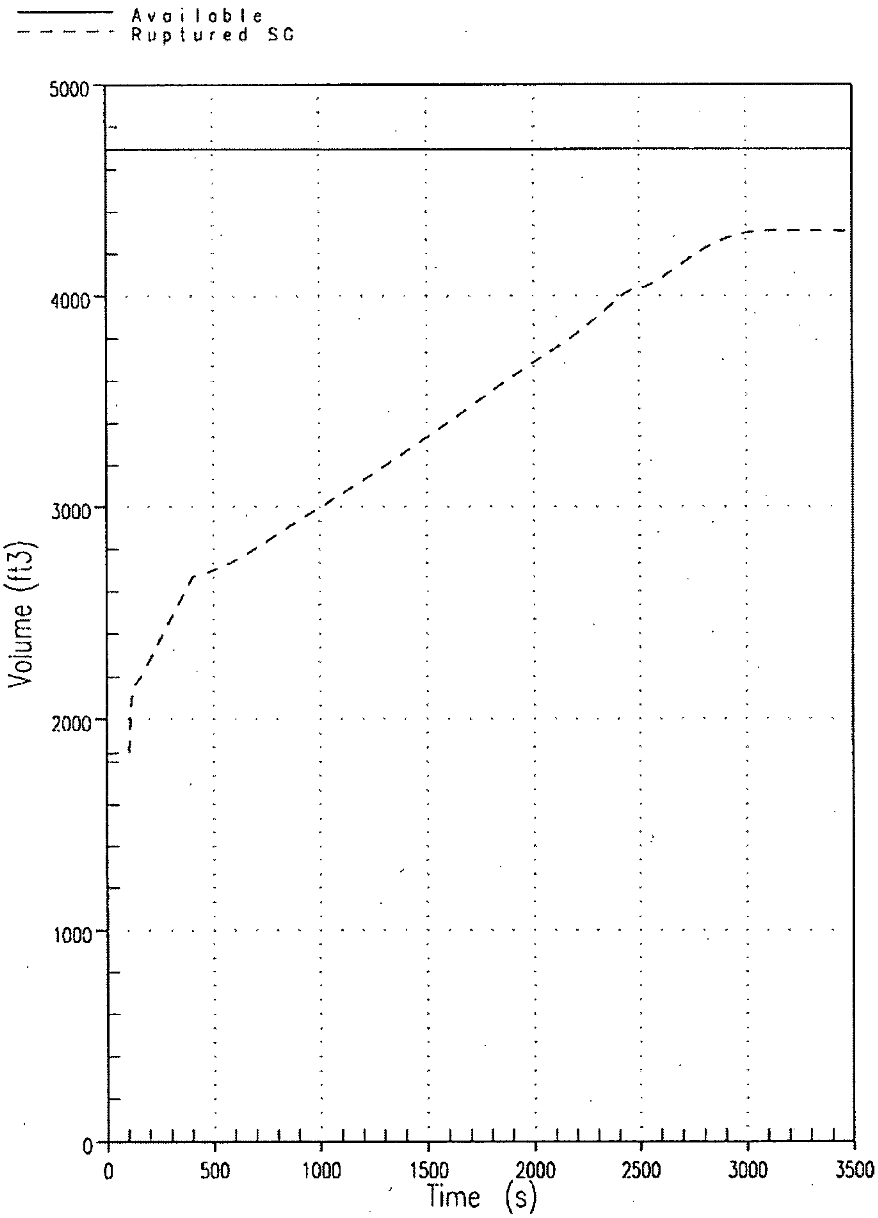


Figure 3: Ruptured Steam Generator Water Volume (EPU)



SRXB-1.2: For the revised margin to overfill/overfill analysis, provide a table comparing analytic assumptions used in WCAP-10698 to those used in the Turkey Point analyses, and justify any differences.

Table 3 provides a comparison of the analytical assumptions used in WCAP-10698-P-A to those used in the Turkey Point analyses.

Table 3 Comparison of WCAP-10698-P-A Modeling to the Revised Analysis Assumptions

Parameter	WCAP-10698 Model	PTN Revised SGTR MTO Analysis
	Direction of Conservatism	**EPU**
Initial Conditions		
Power [1]	Full Power (Nominal + Uncertainty)	Full Power (Nominal + Uncertainty)
RCS Pressure	Minimum	Minimum
Pressurizer Water Level	Maximum	Maximum
SG Secondary Mass	Maximum	Maximum
Break Location	Cold-leg	Cold-leg
Offsite Power Availability		
Offsite Power	Loss of Offsite Power (LOOP)	Loss of Offsite Power (LOOP)
Protection Setpoints and Errors		
Reactor Trip Delay	Minimum	Minimum
Turbine Trip Delay	Minimum	Minimum
SG Relief or Safety Valve Pressure Setpoint	Minimum (PORV)	Minimum (PORV)
Pressurizer Pressure Trip Setpoint	Maximum	Maximum
Pressurizer Pressure SI Setpoint	Maximum	Maximum
Safeguards Capacity		
SI Flow Rate	Maximum	Maximum
AFW Flow Rate (isolation on operator action time)	Maximum	Maximum
AFW System Delay	Minimum	Minimum
AFW Temperature	Maximum	Minimum[2]
Control. Systems		
CVS Operation, PZR Heater Control	Not Operating	Not Operating
Turbine Runback Mass Penalty	Included	Not Included[3]
RCP Running	Not Operating	Not Operating
Decay Heat		
Decay Heat	Maximum	ANS 1979-2σ[2]
Single Failure		
Single Failure	Included	Not Included, consistent with current licensing basis (CLB)

(1) Consistent with the discussion of power in WCAP-10698-P-A, the initial steam generator mass is more conservatively calculated without inclusion of the initial power uncertainty since it results in a higher mass.

(2) For this revised analysis, the 1979 American Nuclear Society (ANS) decay heat model minus 2σ uncertainty is used. Plant specific sensitivities performed to address the NSAL-07-11 issue regarding the use of a higher decay heat uncertainty confirmed that the use of the 1979-2σ decay heat is conservative compared to the 1971+20% ANS decay heat model specified by the methodology of WCAP-10698-P-A. Additionally plant-specific sensitivities for Turkey Point concluded that it is more conservative (i.e., less margin to overfill) to model AFW temperature differently than prescribed by WCAP-10698-P-A.

(3) There is no automatic OTΔT turbine runback system at Turkey Point, and thus no penalty is included. This is an acceptable deviation from the WCAP-10698-P-A methodology since it incorporates plant-specific configuration.

SRXB-1.3: For the SGTR analyses, provide a list of systems, components, and instruments that are credited for accident mitigation in the plant-specific EOPs. Specify whether each component is safety grade, consistent with Requirement (4) of the NRC staff SER approving WCAP-10698.

Information pertaining to credited systems, components, and instruments is presented in the following table. Equipment specific to Unit 3 is shown, but identical equipment is available for Unit 4. Any single component is shown in the table only once, even though some components are relied upon several times throughout the EOPs. The list presents equipment which is specifically utilized in the EOP for mitigating a SGTR event, to include terminating the release from the ruptured steam generator, stopping primary-to-secondary leakage, and restoring RCS pressure, temperature, and inventory control.

Equipment/Tag	Description	Safety Related or Quality Related
3P215A/B	SI Pumps	SR
MOV-3-843A/B	SI Cold Leg Injection Iso Valves	SR
FT-3-943	SI Cold Leg Injection Flow Indication	SR
PT-3-455/456/457	Pressurizer Pressure Indication	SR
EDG 3K4A / B	Emergency Diesel Generators	SR
P2A / P2B / P2C	AFW Pumps A, B, and C	SR
MOV-3-1403/1404/1405	MS Isol to AFW Pumps	SR
CV-3-2816/2817/2818 CV-3-2831/2832/2833	AFW Flow Control Valves	SR(1)
FT-3-1401A/B;1457A/B; 1458A/B	AFW Flow Indication	SR
LT-3-474/475/476 LT-3-484/485/486 LT-3-494/495/496	S/G Narrow Range Level Indication	SR
TR-3-410	RCS Cold Leg Indicator/Recorder	SR
PCV-3-455C/456	Pressurizer PORVs	SR(2)

PT-3-474/475/476 PT-3-484/485/486 PT-3-494/495/496	S/G Pressure Indication	SR
RD-3-15	SJAE Radiation Monitor	QR(3)
RD-3-19	SGBD Radiation Monitor	QR(3)
RAD-3-6417	SJAE SPING Radiation Monitor	QR(3)
FT-3-474/475 FT-3-484/485 FT-3-494/495	S/G Steam Flow Indication	SR
CV-3-1606/1607/1608	MSL Steam Dumps to Atmosphere	SR(4)
3CM / 3CD	Instrument Air Compressors	NNS
CV-3-6275A/B/C	SG Blowdown Isolation Valves	SR
POV-3-2604/2605/2606	MSIVs	SR
MOV-3-1400/1401/1402	MS bypass valves	SR
MOV-3-1425/1426/1427	S/G C Sample Line Isolation Valves	SR
TE-3-#E (Various)	Core exit thermocouples	SR
CV-3-2827/2828 CV-3-2829/2930	Steam Dump to Condenser Valves	QR(5)
MOV-3-535/536	Pressurizer PORV block valve	SR
PT-3-403	RCS Wide Range Pressure Indication	SR
FT-3-605	Flow Indicator for RHR	SR
3P201A/B/C	Charging Pumps	SR
PCV-3-455A/B	Pressurizer Spray Valves	SR(6)
CV-3-311	Auxiliary Spray Valves	SR(6)
LT-3-460/461/462	PRZ Level Indicator	SR
MOV-3-865A/B/C	SI Accumulator Isolation MOVs	SR

Table Notes:

(1) Equipment is safety-related with a dedicated, safety-related nitrogen backup supply.

(2) Equipment is safety-related with a dedicated, quality-related nitrogen backup supply.

(3) Operators are directed to perform steam line surveys and monitor steam generator level indications to identify the affected steam generator, in addition to steam generator sampling. Delays associated with sampling will not delay the performance of mitigating actions, since steam and feedwater flow mismatch, level indications, and radiation surveys will provide clear indication of the affected steam generator.

(4) Equipment is safety related with nitrogen backup. Nitrogen for the control signal is from a dedicated, quality-related bottled source. Nitrogen for motive force on the valve operator is from the plant nitrogen system via quality-related supply piping.

(5) Equipment is quality-related with non-nuclear safety (NNS) instrument air supplied to the operators.

(6) Equipment is safety-related as an RCS pressure boundary, but the operator is supplied by NNS instrument air.

SRXB-1.4: Under assumed loss of offsite power (LOOP) conditions, address the functionality of each atmospheric dump valve (ADV). Discuss what, if any, mitigating function the ADV provides and its capability to perform that function under the assumed LOOP conditions.

An SGTR event is mitigated by isolating the affected steam generator, cooling down the RCS to maintain adequate subcooling, and depressurizing the RCS to eliminate reactor coolant leakage through the tube rupture and maintain RCS inventory. When offsite power is available, the steam dump to condenser valves are used to dump steam from the intact steam generators to the condenser to cooldown the RCS. However, during a LOOP, main feedwater and condensate systems are unavailable; instead, the ADVs on the intact steam generators are used for cooldown, in conjunction with the turbine-driven auxiliary feedwater pumps or the diesel-driven standby steam generator feedwater pump.

The Turkey Point ADVs are air-operated angle globe valves, configured as air-to-open / spring-to-close. One ADV is provided on each steam header upstream of the main steam isolation valves, totaling three ADVs per Unit.

Air for the ADV pneumatic operators is normally supplied by the instrument air system. Each Unit is equipped with one electric motor-driven instrument air compressor, and one diesel-driven air compressor, for a total of four compressors. The two Units' instrument air systems are normally cross connected, and any one of the four compressors alone can supply the combined instrument air load for both Units operating simultaneously.

A LOOP to either Unit will de-energize that Unit's motor-driven instrument air compressor. However, the associated diesel-driven compressor will automatically start on a loss of power to maintain continuity of instrument air service. In addition, the affected Unit's instrument air dryer is automatically sequenced onto the emergency diesel generators during LOOP conditions. With this arrangement, instrument air is automatically and immediately restored during a LOOP without operator action.

The ADVs are controlled from panel-mounted hand/auto digital controllers in the main control room. A separate controller is provided for each ADV. In the automatic mode, the controller issues a pneumatic valve position signal based on a comparison between the operator-selected setpoint and a digital non-safety related main steam pressure signal. In the manual mode, the operator adjusts the digital controller to directly manipulate the pneumatic valve position signal. To generate the pneumatic valve position signal, the controller receives an air supply that is auctioneered between instrument air (normal) and a dedicated bottled nitrogen source (backup).

Each controller receives electrical power from vital inverter-backed 120 VAC panels. Use of diverse vital AC power panels assures that for any failure event, at least two steam dump controllers will always be available. The use of vital AC power ensures controller availability during LOOP conditions. A closed-position limit switch is installed on each ADV to provide Control Room operators with closed/not-closed position indication via the plant's Digital Control System.

With a LOOP, a total loss of instrument air would require the failure of both diesel-driven compressors. In that event, motive force for the valves' operators is backed up through reducers from the plant's nitrogen system (about 80 psig). Separately, the 3 to 15 psig pneumatic control signal to the ADV positioners would be supplied from a dedicated nitrogen bottle station, where one bottle has sufficient capacity to allow continuous operation of one Unit's ADVs for 3 hours, and subsequently to maintain their position for 8.5 hrs. One bottle is normally valved in, and one additional cylinder is added to act as a common source for both units should extended steady state operation be required. This arrangement of redundant air supplies, along with the inverter-backed controller power supplies, ensures the reliability of each ADV.

SRXB-1.5: Identify any new operator actions credited in the revised margin to overfill/overfill analysis.

There are no new operator actions credited in the above analysis to that provided in Reference 1.

SRXB-1.6: Section 2.8.5.6.3 describes a more refined downcomer model. Provide the following specific information concerning the downcomer model:

a. **Provide a detailed description and diagram of the downcomer nodalization, including both fluid and heat structures.**
b. **Identify the sources of heat modeled in the downcomer.**

Parts a and b of this RAI are addressed together since they are related questions.

The noding diagram for Turkey Point Units 3 and 4 with nine downcomer channel stacks is presented as Figure 4. The numbers enclosed in squares represent channel numbers. Channels are used to make vertical connections in the vessel model. The numbers enclosed in circles represent gap numbers. Gaps are used to make lateral connections in the vessel model. The gap numbers which have a horizontal arrow through them connect the channels shown at the start and end of the horizontal arrow. The gaps which have a diagonal arrow through them have a corresponding numbered gap shown elsewhere on the noding diagram. These gaps connect the channels with the matching gap numbers shown.

The downcomer channels are modeled with the long channel stacks shown on the outer portion of the noding diagram (Figure 4).

Cross-sections of the vessel noding at each section elevation are presented as Figures 5, 6, and 7. The cold legs are connected to channels 30, 31, and 32 and the hot legs are connected to channels 37, 38, and 39 in Section 6 as shown in Figure 6.

The metal structures connected to the downcomer which serve as a heat source during a LBLOCA are shown in Figure 8. Only the downcomer channels are shown in this figure, and the gaps are omitted for clarity. The numbers in squares are again the channel numbers, and the unheated conductors are designated with diamonds. The structure to channel connections and a description of the structures are contained in Table 4.

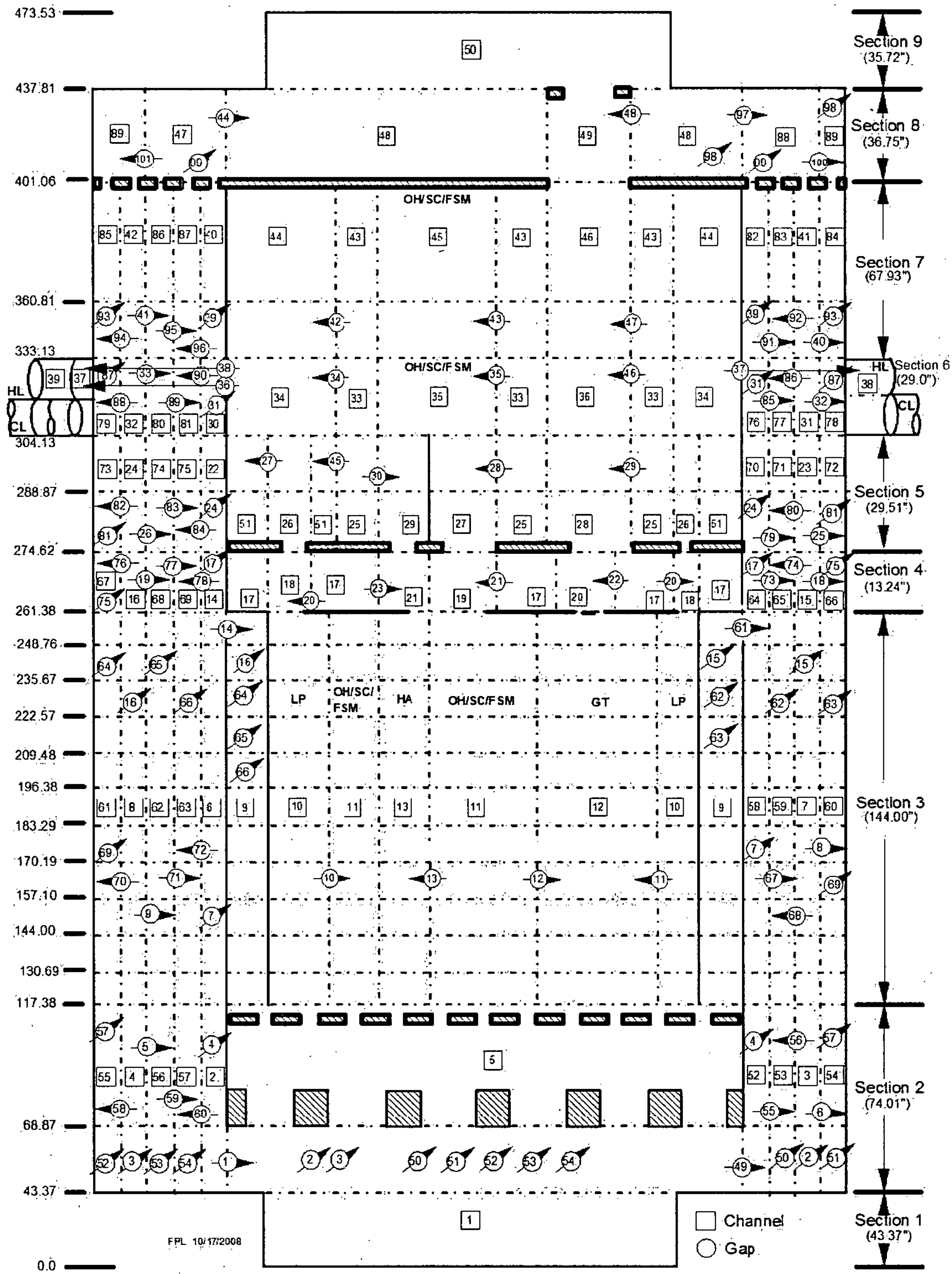


Figure 4: Turkey Point Units 3 and 4 Vessel Noding Diagram for the Nine Downcomer Channel Stack Model

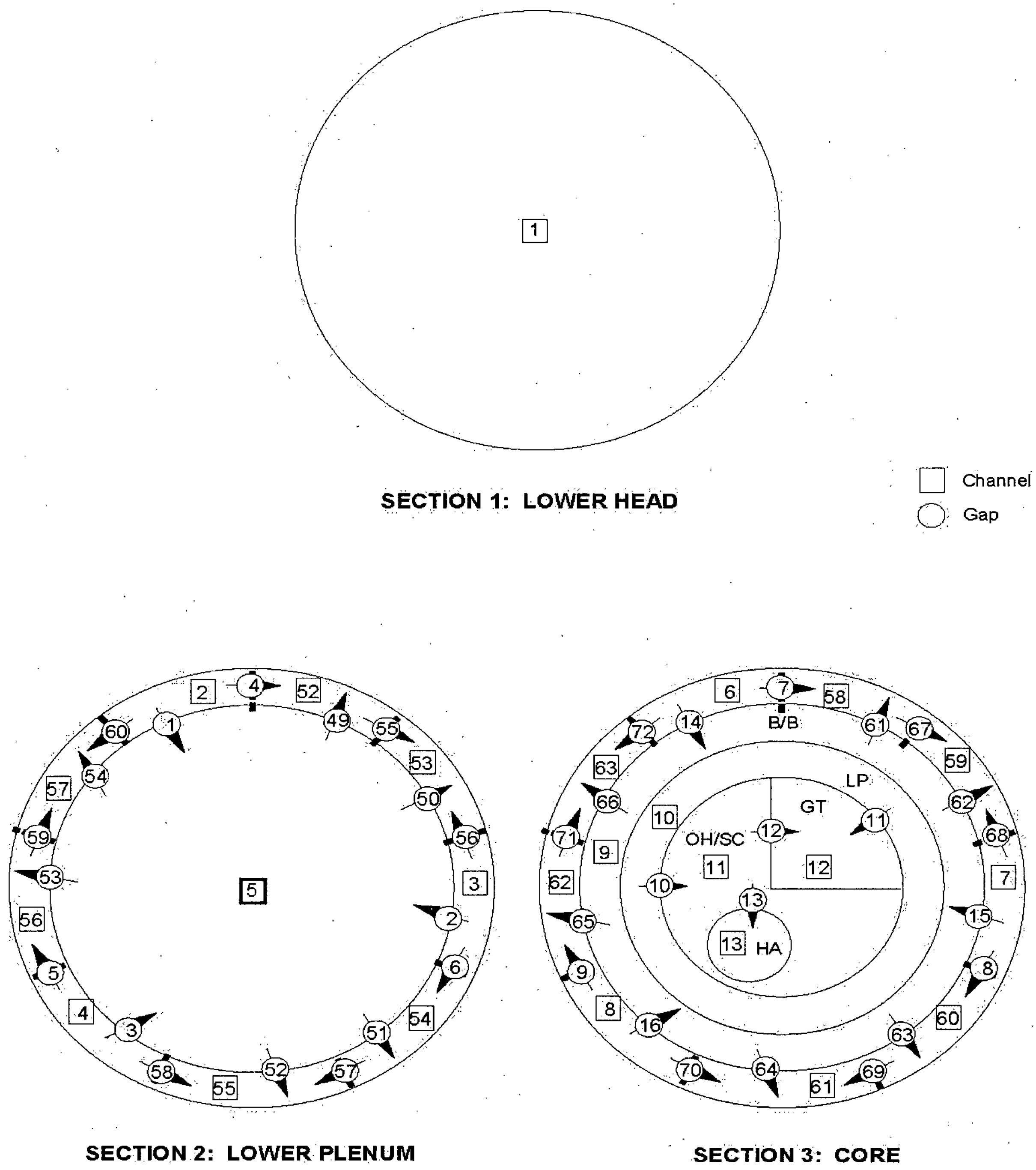


Figure 5: Turkey Point Units 3 and 4 Cross-Section Diagram for Vessel Sections 1, 2, and 3

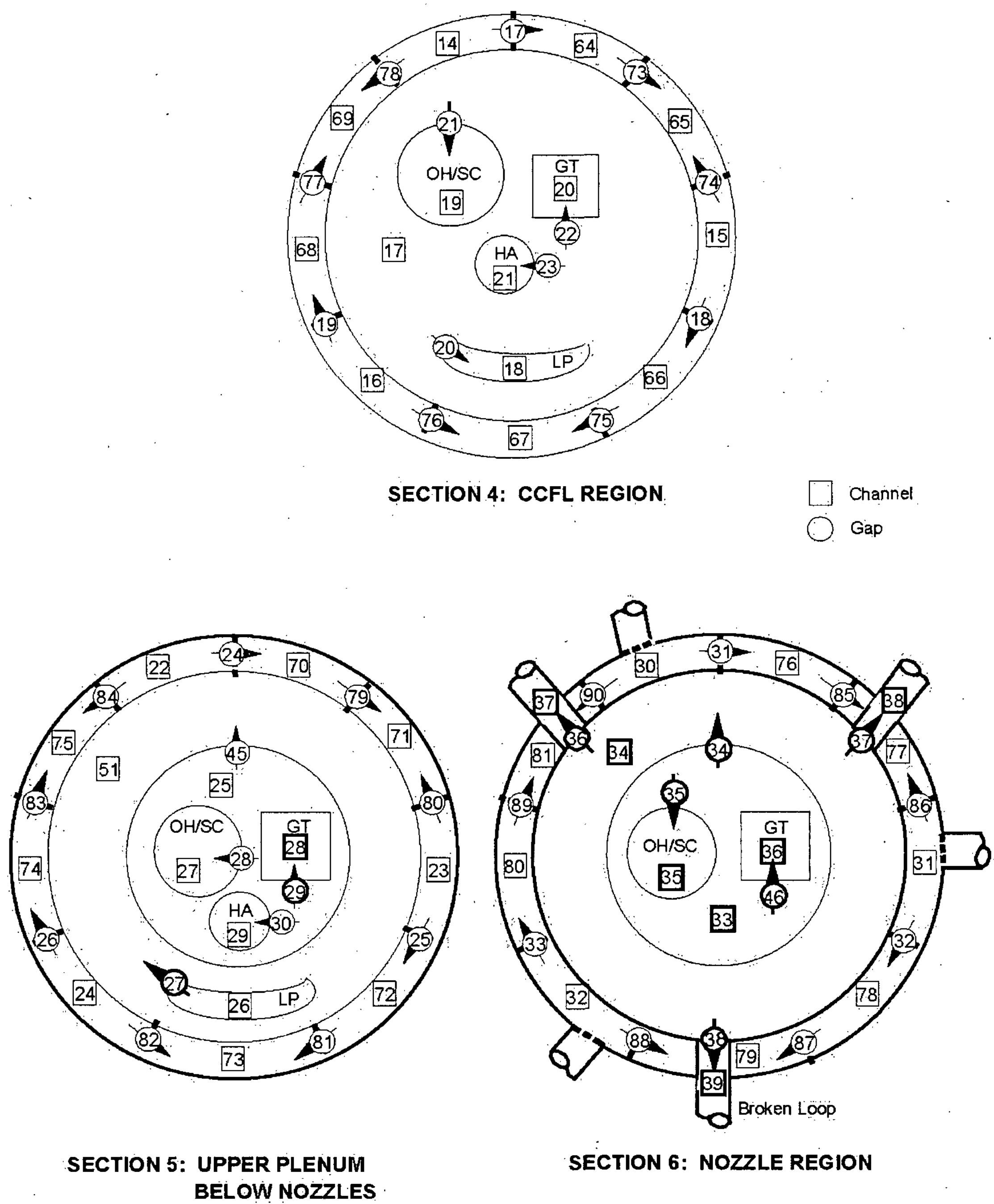
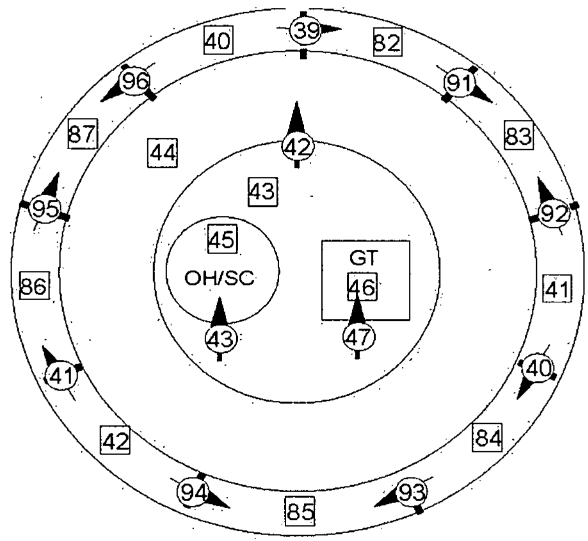
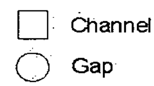
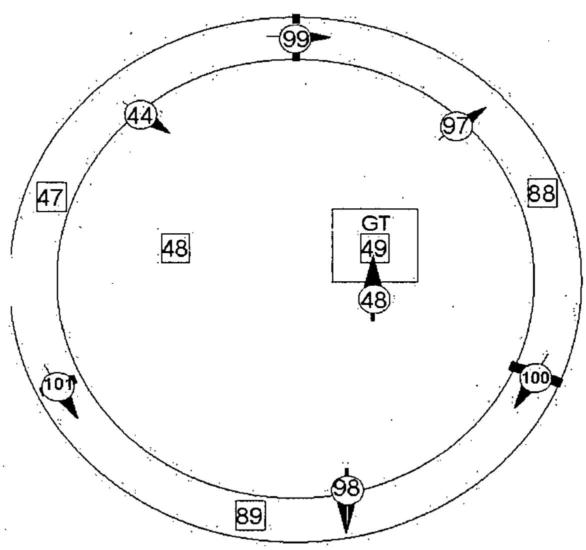


Figure 6: Turkey Point Units 3 and 4 Cross-Section Diagram for Vessel Sections 4, 5, and 6

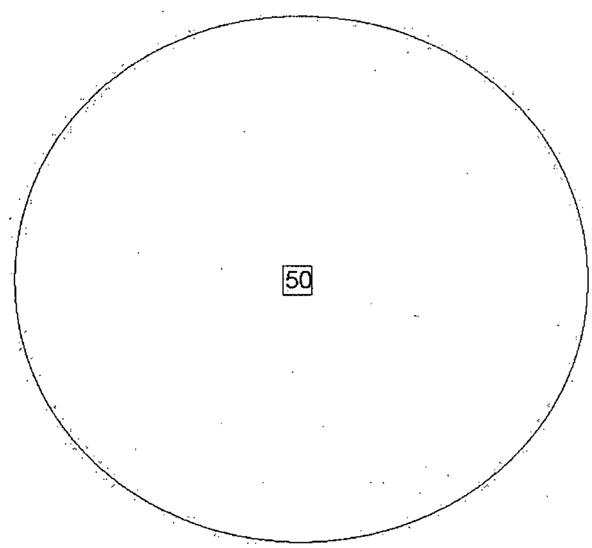


SECTION 7: UPPER PLENUM ABOVE NOZZLES





SECTION 8: UPPER HEAD UP TO TOP OF GUIDE TUBES



SECTION 9: UPPER HEAD ABOVE GUIDE TUBES

Figure 7: Turkey Point Units 3 and 4 Cross-Section Diagram for Vessel Sections 7, 8, and 9

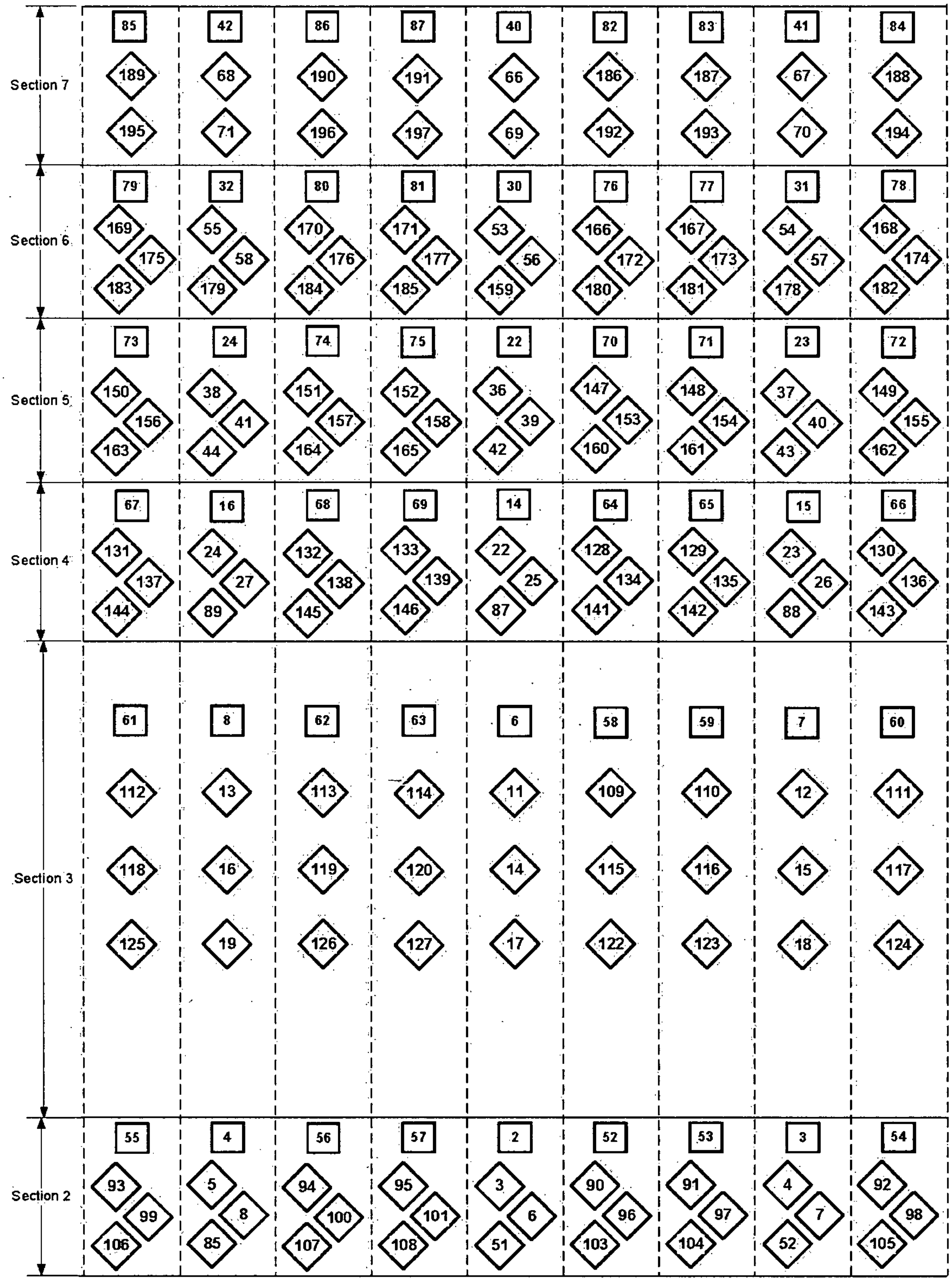


Figure 8: Metal Structures Connected to the Downcomer in the PTN Units 3 and 4 Vessel Model

Table 4 (Page 1 of 4): Metal Structures connected to the Downcomer in the Turkey Point Units 3 and 4 Vessel Model

Unheated Conductor Number	Downcomer Channel Connected	Description
3	2	One-ninth of vessel wall in Vessel Section 2
4	3	
5	4	
6	2	One-ninth of thermal shield and radial keys in Vessel Section 2
7	3	
8	4	
11	6	One-ninth of vessel wall in Vessel Section 3
12	7	
13	8	
14	6	One-ninth of thermal shield in Vessel Section 3
15	7	
16	8	
17	6	One-ninth of outer half of core barrel in Vessel Section 3
18	7	
19	8	
22	14	One-ninth of vessel wall in Vessel Section 4
23	15	
24	16	
25	14	One-ninth of thermal shield in Vessel Section 4
26	15	
27	16	
36	22	One-ninth of vessel wall in Vessel Section 5
37	23	
38	24	
39	22	One-ninth of outer half of core barrel in Vessel Section 5
40	23	
41	24	
42	22	One-ninth of thermal shield in Vessel Section 5
43	23	
44	24	
51	2	One-ninth of outer half of core barrel in Vessel Section 2
52	3	
53	30	One-ninth of vessel wall in Vessel Section 6
54	31	
55	32	
56	30	One-ninth of outer half of core barrel in Vessel Section 6
57	31	
58	32	
66	40	One-ninth of vessel wall in Vessel Section 7
67	41	
68	42	

Table 4 (Page 2 of 4): Metal Structures connected to the Downcomer in the Turkey Point Units 3 and 4 Vessel Model

69	40	One-ninth of outer half of core barrel in Vessel Section 7
70	41	
71	42	
85	4	One-ninth of outer half of core barrel in Vessel Section 2
87	14	One-ninth of outer half of core barrel in Vessel Section 4
88	15	
89	16	
90	52	One-ninth of vessel wall in Vessel Section 2
91	53	
92	54	
93	55	
94	56	
95	57	
96	52	One-ninth of outer half of core barrel in Vessel Section 2
97	53	
98	54	
99	55	
100	56	
101	57	
103	52	One-ninth of thermal shield and radial keys in Vessel Section 2
104	53	
105	54	
106	55	
107	56	
108	57	
109	58	One-ninth of vessel wall in Vessel Section 3
110	59	
111	60	
112	61	
113	62	
114	63	
115	58	One-ninth of outer half of core barrel in Vessel Section 3
116	59	
117	60	
118	61	
119	62	
120	63	
122	58	One-ninth of thermal shield in Vessel Section 3
123	59	
124	60	
125	61	
126	62	
127	63	

Table 4 (Page 3 of 4): Metal Structures connected to the Downcomer in the Turkey Point Units 3 and 4 Vessel Model

128	64	One-ninth of vessel wall in Vessel Section 4
129	65	
130	66	
131	67	
132	68	
133	69	
134	64	One-ninth of outer half of core barrel in Vessel Section 4
135	65	
136	66	
137	67	
138	68	
139	69	
141	64	One-ninth of thermal shield in Vessel Section 4
142	65	
143	66	
144	67	
145	68	
146	69	
147	70	One-ninth of vessel wall in Vessel Section 5
148	71	
149	72	
150	73	
151	74	
152	75	
153	70	One-ninth of outer half of core barrel in Vessel Section 5
154	71	
155	72	
156	73	
157	74	
158	75	
159	30	One-ninth of the core barrel ring in Vessel Section 6
160	70	One-ninth of thermal shield in Vessel Section 5
161	71	
162	72	
163	73	
164	74	
165	75	
166	76	One-ninth of vessel wall in Vessel Section 6
167	77	
168	78	
169	79	
170	80	
171	81	

Table 4 (Page 4 of 4): Metal Structures connected to the Downcomer in the Turkey Point Units 3 and 4 Vessel Model

172	76	One-ninth of outer half of core barrel in Vessel Section 6
173	77	
174	78	
175	79	
176	80	
177	81	
178	31	One-ninth of the core barrel ring in Vessel Section 6
179	32	
180	76	
181	77	
182	78	
183	79	
184	80	
185	81	
186	82	One-ninth of vessel wall in Vessel Section 7
187	83	
188	84	
189	85	
190	86	
191	87	
192	82	One-ninth of outer half of core barrel in Vessel Section 7
193	83	
194	84	
195	85	
196	86	
197	87	

c. Discuss how subcooled boiling in the downcomer is modeled.

The treatment of subcooled boiling in the downcomer for the nine downcomer channel stack model is the same as for the three downcomer channel stack model. Subcooled boiling in the downcomer is calculated using the Chen (Reference 4) correlation. While the Chen correlation was developed for saturated boiling, it can be extended into the subcooled region. The Chen correlation superimposes a forced convective and a nucleate boiling component. Moles and Shaw (Reference 6) compared the Chen correlation to boiling data for several fluids and reported satisfactory agreement for low to moderate subcooling.

During subcooled boiling vapor generation occurs and a significant void fraction may exist despite the presence of subcooled water. In this regime, three processes are of interest relative to the downcomer region:

1. forced convection to the liquid,
2. vapor generation at the wall, and
3. condensation near the wall.

Forced convection to the liquid is treated by the forced convective component of the Chen correlation to determine the heat input into the liquid. The nucleate boiling component of the Chen correlation defines the amount of heat available to cause vapor generation at the wall. The near-wall condensation is estimated using the Hancox-Nicoll (Reference 5) correlation for heat flux at the point where all the bubbles generated collapse in the near-wall region.

Please refer to Section 6-2-3 of the CQD (Bajorek et al., Reference 3) for additional information regarding the treatment of subcooled boiling.

SRXB-1.7: By letter dated October 21, 2010, the license amendment request (LAR) states, "As noted in PTN Updated Final Safety Analysis Report (UFSAR), Section 1.3, the General Design Criteria (GDC) used during licensing of the Turkey Point Nuclear Plant predate those provided today in 10 CFR 50, Appendix A. The PTN GDCs were developed based on the 1967 Atomic Energy Commission Proposed General Design Criteria and are addressed in various sections of the UFSAR."

The LAR also identifies, as one of the GDCs in the Turkey Point licensing basis, *PTN GDC-30, Reactivity Holddown Capability: "The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public."* The LAR states that PTN GDC-30 is comparable to the current GDC-27.

However, the 1967 proposed GDC (32 FR 10213) that corresponds to PTN GDC-30 is *"Criterion 30--Reactivity Holddown Capability (Category B).* At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies."

Apparently, the 1967 proposed GDC-30 is more restrictive than PTN GDC-30. Explain and justify the difference between PTN GDC-30 and its basis, and the 1967 proposed GDC-30.

Explain how PTN GDC-30 is considered to be equivalent to the current GDC-27, not the current GDC-26.

PTN's licensing basis was and is based on the proposed AEC GDCs as amended by the Atomic Industrial Forum (AIF).

1967 AEC GDC 30, Reactivity Holddown Capability, states: "At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies."

1967 AIF Reworded AEC GDC 30, Reactivity Holddown Capability, states: "The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public."

10 CFR 50, Appendix A, Criterion 27, Combined reactivity control systems capability, states: "The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained."

The original proposed AEC GDC 30 requirements are more restrictive than those in either the version adopted by Turkey Point or the current 10 CFR 50 Appendix A. The first would require a single reactivity system alone be capable of holding the core subcritical prohibiting return to power under all conditions. The PTN version, described in Section 3.1.2 of the UFSAR, requires that the reactivity systems together be capable of initially making the core subcritical for all credible accident conditions and limit any subsequent return to power.

10 CFR 50, Appendix A, Criterion 26, Reactivity control system redundancy and capability, states: "Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions."

Both of the above GDCs address the capability of the reactivity control systems. GDC 26 addresses the requirements under normal operation and anticipated operational occurrences. It also addresses reactor holddown capability under cold conditions. GDC 27 addresses the requirements under postulated accident conditions, consistent with PTN-GDC 30.

GDC-27 and PTN-GDC-30 provide for appropriate margins in reactivity capability ("with appropriate margins for contingencies" vs. "with appropriate margin for stuck rods", respectively). Both provide for multiple reactivity control systems to satisfy the requirements of the GDC ("The reactivity control systems" for both GDCs). Both have similar success criteria ("limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public" vs. "to assure . . . the capability to cool the core is maintained", respectively).

References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010
2. Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU – Reactor Systems (SRXB) Requests for Additional Information – Round 1", Accession No. ML110460085, February 15, 2011.
3. Bajorek, S. M., et al., March 1998, "Code Qualification Document for Best Estimate LOCA Analysis," Volume 1 Revision 2, and Volumes 2 through 5, Revision 1, WCAP-12945-P-A (Proprietary).
4. Chen, J. C., 1963, "A Correlation for Boiling Heat Transfer to Saturated Fluids in Convective Flow," ASME 63-HT-34.
5. Hancox, W. T. and Nicoll, W. B., 1971, "A General Technique for the Prediction of Void Distributions in Nonsteady Two-Phase Forced Convection," <u>Int. J. Heat and Mass Transfer</u>, Vol. 14.
6. Moles, F. D., and Shaw, J. F. G., 1972, "Boiling Heat Transfer to Subcooled Liquids Under Conditions of Forced Convection," <u>Trans. Inst. Chem. Eng.</u>, Vol. 50.