APPENDIX 14A

TURKEY POINT PLANT UNIT 3
Core Operating Limits Report (COLR)

C30

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C31

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1.0 SUMMARY

The Core Operating Limits Report (COLR) for Unit 3 Cycle 31 is provided in Appendix A. C31

Table 14A-1

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Table 14A-2

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Table 14A-3

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Table 14A-4

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Figure 14A-1

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Figure 14A-2

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Appendix A

Turkey Point Unit 3 Cycle 31
Core Operating Limits Report (COLR)

C31

Turkey Point Unit 3 Cycle 31
Core Operating Limits Report (COLR)

C31

1.0 INTRODUCTION

This Core Operating Limits Report for Turkey Point Unit 3 Cycle 31 has been prepared in accordance with the requirements of Technical Specification 6.9.1.7. C31

The Technical Specifications (TS) affected by this report are listed below with the section and page for each one of the TS addressed in this COLR document.

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2.0 Operating Limits

The cycle-specific parameter limits for the specifications listed in the Introduction are presented below and listed sequentially by Technical Specification (TS). These limits have been developed using the NRC-approved methodologies specified in TS 6.9.1.7.

2.1 Reactor Core Safety Limits – Three Loops in Operation (TS 2.1.1)

- **Figure A1** (page 14A-A7) — In Modes 1 and 2, the combination of Thermal Power, reactor coolant system highest loop average temperature and pressurizer pressure shall not exceed the limits in Figure A1.

2.2 Reactor Trip System Instrumentation Setpoints (TS 2.2.1)

<u>NOTE 1 on TS Table 2.2-1 Overtemperature ΔT</u>

- **$\tau 1$ = 0s, $\tau 2$ = 0s** — Lead/Lag compensator on measured ΔT
- **τ_3 = 2s** — Lag compensator on measured ΔT
- **K_1 = 1.31**
- **K_2 = 0.023/°F**
- **τ_4 = 25s, τ_5 = 3s** — Time constants utilized in the lead-lag compensator for T_{avg}
- **τ_6 = 2s** — Lag compensator on measured T_{avg}
- **T′ ≤ 583.0 °F** — Indicated Loop T_{avg} at RATED THERMAL POWER
- **K_3 = 0.00116/psi**
- **P’ ≥ 2235 psig** — Nominal RCS operating pressure
- **$f_1(\Delta I)$ = 0 for $q_t - q_b$ between – 18% and + 7%**.

For each percent that the magnitude of $q_t - q_b$ exceeds – 18%, the ΔT Trip Setpoint shall be automatically reduced by 3.51% of its value at RATED THERMAL POWER; and

For each percent that the magnitude of $q_t - q_b$ exceeds + 7%, the ΔT Trip Setpoint shall be automatically reduced by 2.37% of its value at RATED THERMAL POWER.

Where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

NOTE 2 on TS Table 2.2-1 Overtemperature ΔT

The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.2% ΔT span for the Pressurizer Pressure channel, and 0.4% ΔT span for the f(ΔI) channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

NOTE 3 on TS Table 2.2-1 Overpower ΔT

- **K_4 = 1.10**

- **$K_5 \geq 0.0/^\circ F$**	For increasing average temperature
- **$K_5 = 0.0/^\circ F$**	For decreasing average temperature
- **$\tau_7 \geq 0$ s**	Time constants utilized in the lead-lag compensator for T_{avg}
- **$K_6 = 0.0016/^\circ F$**	For T > T"
- **$K_6 = 0.0$**	For T $\leq$ T"
- **T" $\leq 583.0^\circ F$**	Indicated Loop T_{avg} at RATED THERMAL POWER
- **f_2 (ΔI) = 0**	For all ΔI

NOTE 4 on TS Table 2.2-1 Overpower ΔT

The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

2.3 Shutdown Margin Limit for MODES 1, 2, 3 and 4 (TS 3.1.1.1)

- **Figure A2** (page 14A-A8)

2.4 Shutdown Margin Limit for MODE 5 (TS 3.1.1.2)

- **≥ 1.77% $\Delta k/k$**

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2.5 Moderator temperature coefficient (MTC) (TS 3.1.1.3)

- **$\leq$+ 5.0 x 10^{-5} Δk/k/°F** — BOL, HZP, ARO and from HZP to 70% Rated Thermal Power (RTP)

- **From 70% RTP to 100% RTP the MTC decreasing linearly from $\leq$ + 5.0 x 10^{-5} Δk/k/°F to $\leq$ 0.0 x 10^{-5} Δk/k/°F**

- **Less negative than - 41.0 x 10^{-5} Δk/k/°F** — EOL, RTP, ARO

2.6 Moderator temperature coefficient (MTC) Surveillance at 300 ppm (TS 4.1.1.3)

- **Less negative than - 35.0 x 10^{-5} Δk/k/°F (-35 pcm/°F)** — Within 7 EFPD of reaching equilibrium boron concentration of 300 ppm.

The Revised Predicted near - EOL 300 ppm MTC shall be calculated using the algorithm contained in WCAP-13749-P-A:

Revised predicted MTC = Predicted MTC + AFD Correction - 3 pcm/°F

If the Revised Predicted MTC is less negative than the SR 4.1.1.3.b 300 ppm surveillance limit and all the benchmark criteria contained in the surveillance procedure are met, then an MTC measurement in accordance with SR 4.1.1.3.b is not required to be performed.

The neutronics methods used with WCAP-13749-P-A are those described in WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport CODE PARAGON," August 2004. C31

2.7 Analog Rod Position Indication System (TS 3.1.3.2)

- **Figure A3** (page 14A-A9) — The All Rods Out (ARO) position for all shutdown Banks and Control Banks is defined to be 229 steps withdrawn. C31

2.8 Control Rod Insertion Limits (TS 3.1.3.6)

- **Figure A3** (page 14A-A9) — The control rod banks shall be limited in physical insertion as specified in Figure A3 for ARO =229 steps withdrawn. C31

2.9 Axial Flux Difference (TS 3.2.1)

- **Figure A4** (page 14A-A10)

2.10 Heat Flux Hot Channel Factor $F_Q(Z)$ (TS 3.2.2)

- **$[FQ]^L$ = 2.30**
- **K(z) = 1.0** For 0' ≤ z ≤ 12' where z is core height in ft

2.11 Nuclear Enthalpy Rise Hot Channel Factor (TS 3.2.3)

- **$F_{\Delta H}^{RTP}$ = 1.600** **$PF_{\Delta H}$ = 0.3**

2.12 DNB Parameters (TS 3.2.5)

- **RCS Tavg ≤ 585.0 °F**
- **Pressurizer Pressure ≥ 2204 psig**

Figure A1

Reactor Core Safety Limit - Three Loops in Operation

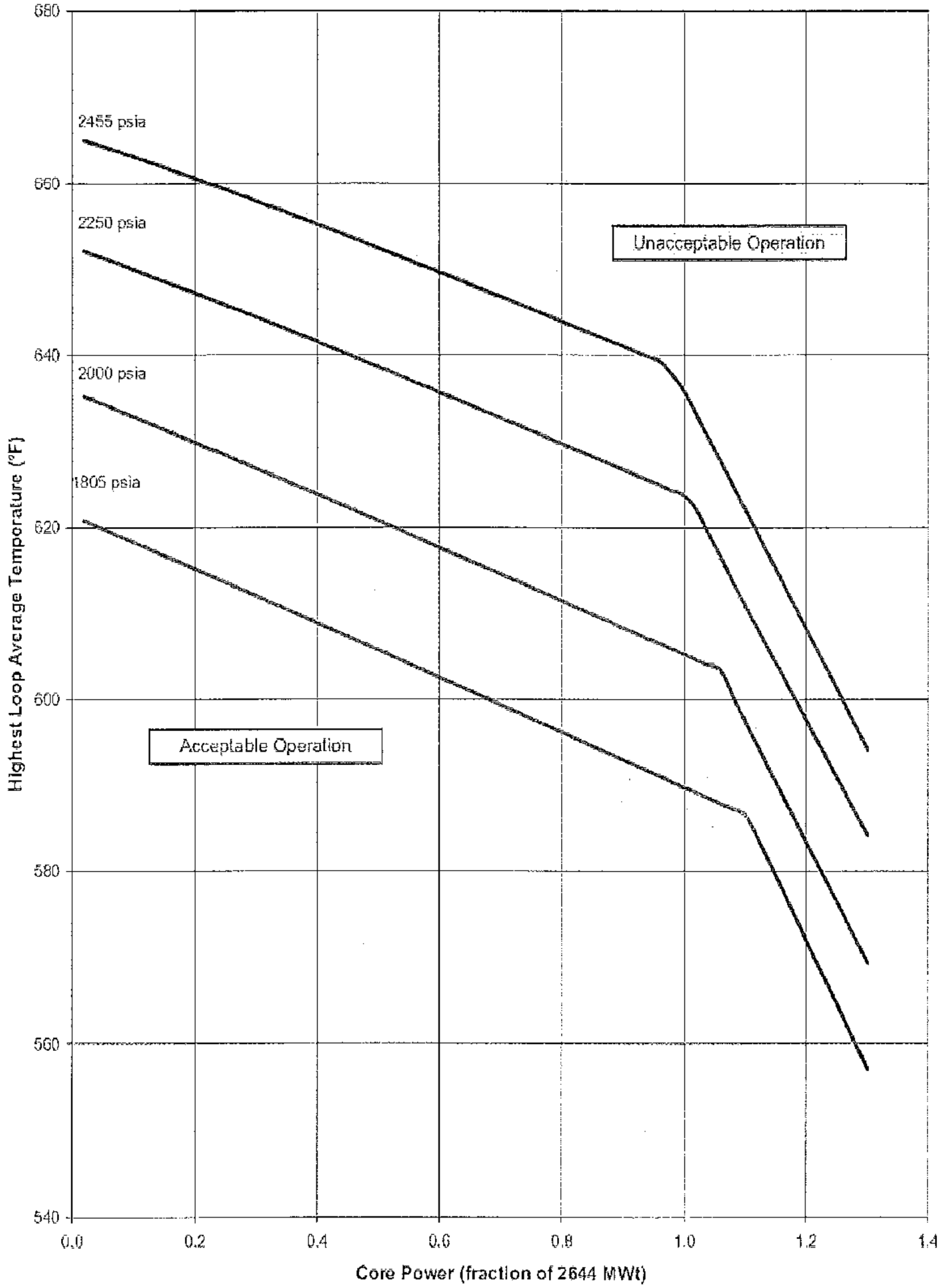
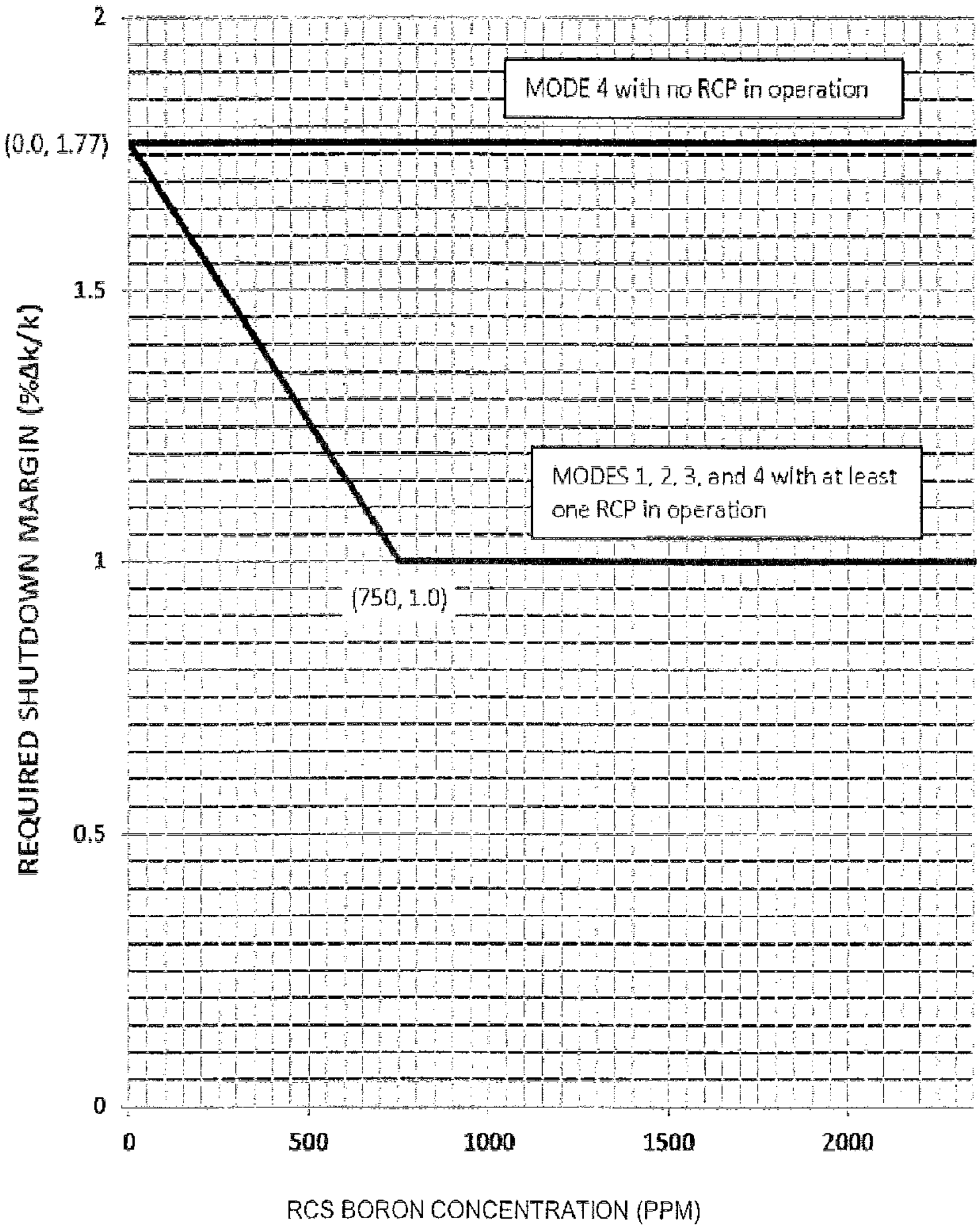


Figure A2

Required Shutdown Margin vs Reactor Coolant Boron Concentration



 Revised 03/11/2016

FIGURE A3

Turkey Point Unit 3 Cycle 31 Rod Insertion Limits vs Thermal Power
ARO = 229 Steps Withdrawn, Overlap = 101 Steps

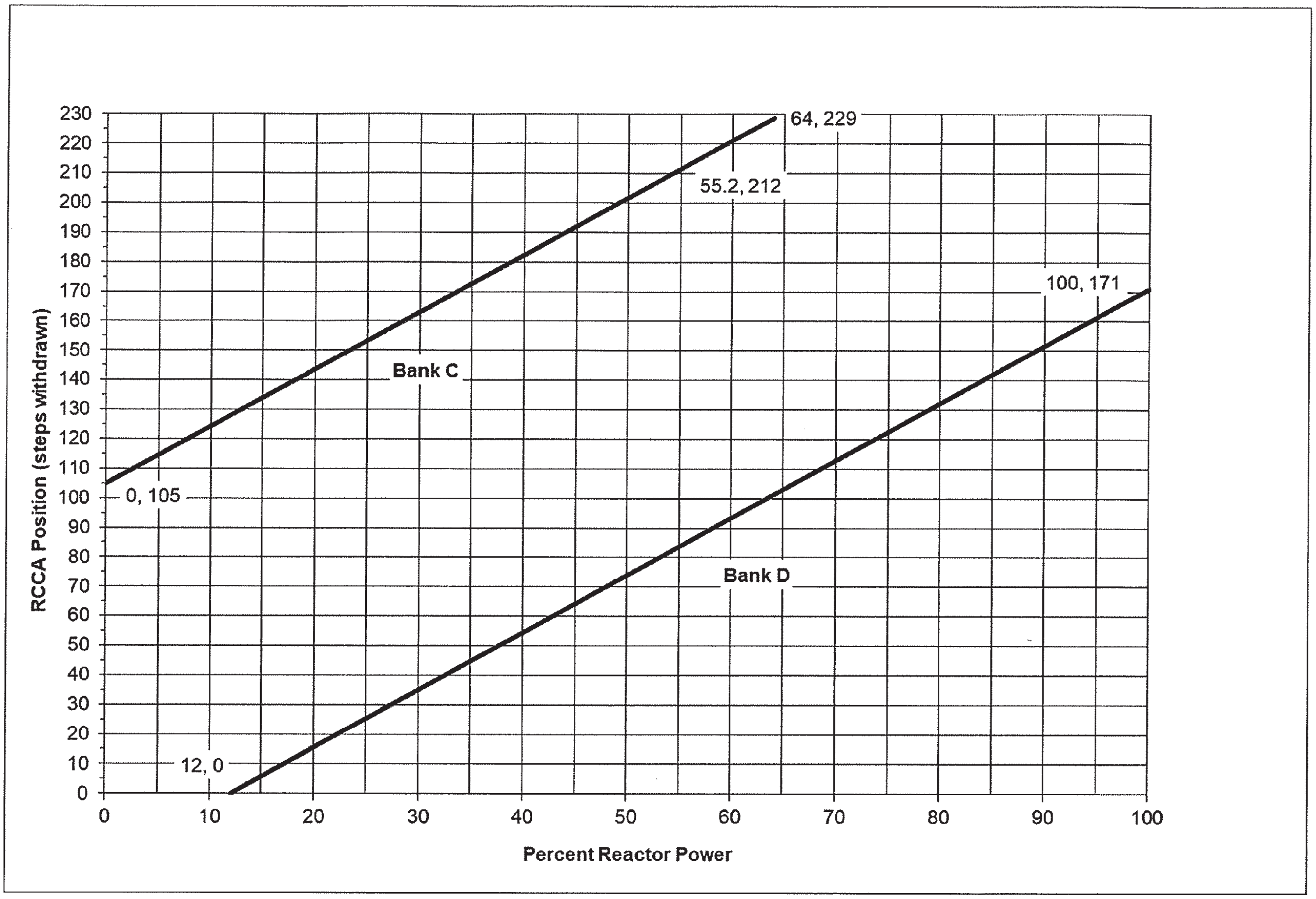


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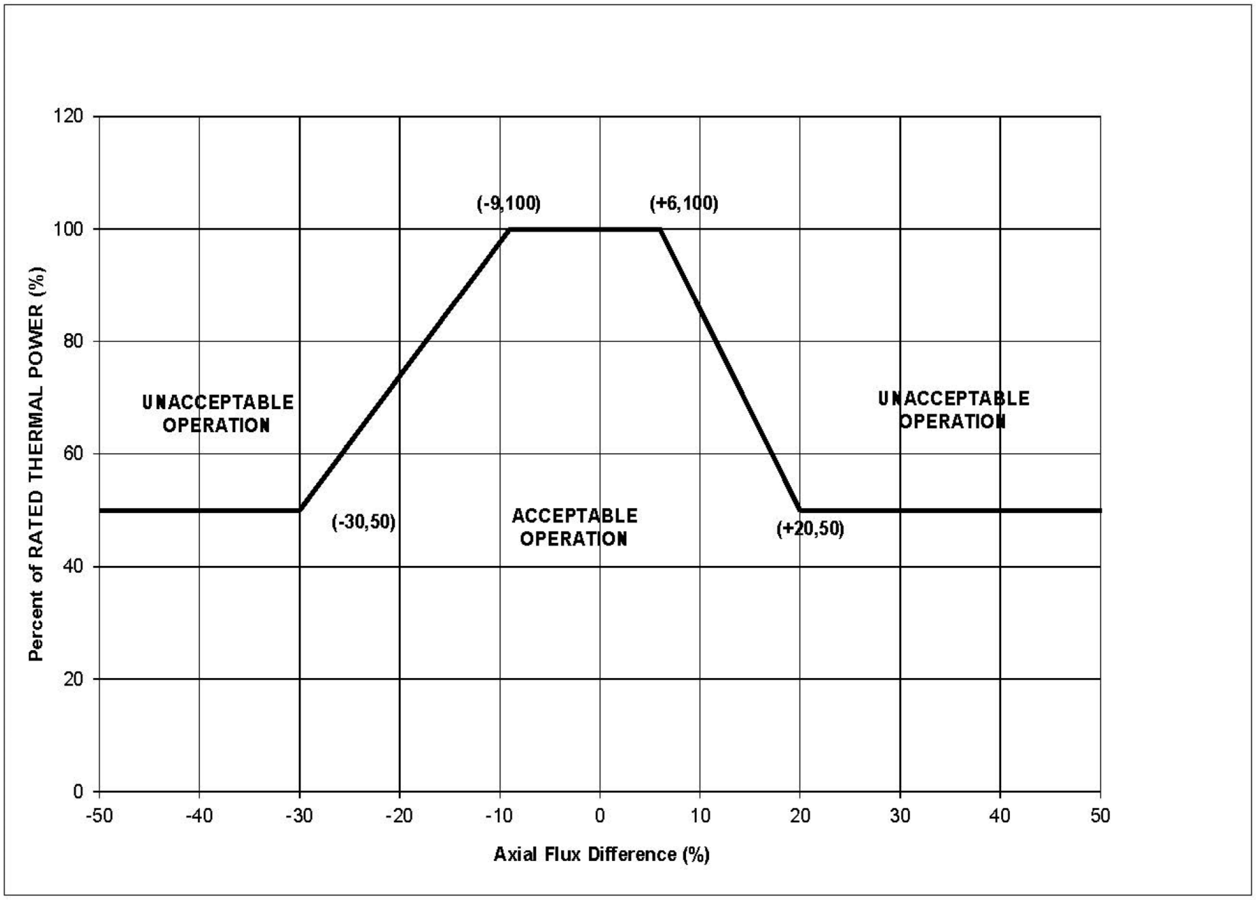
FIGURE A4

Axial Flux Difference as a Function of Rated Thermal Power
Turkey Point Unit 3 Cycle 31

C31



APPENDIX 14B

TURKEY POINT PLANT UNIT 4
Core Operating Limits Report (COLR) C30

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1.0 SUMMARY

The Core Operating Limits Report (COLR) for Unit 4 Cycle 32 is provided in Appendix A. C31

Table 14.B-1

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Table 14B-2

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Table 14B-3

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Figure 14B-1

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Figure 14B-2

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Appendix A

Turkey Point Unit 4 Cycle 32
Core Operating Limits Report (COLR) C31

Turkey Point Unit 4 Cycle 32
Core Operating Limits Report

C31

1.0 Introduction

This Core Operating Limits Report for Turkey Point Unit 4 Cycle 32 has been prepared in accordance with the requirements of Technical Specification 6.9.1.7. C31

The Technical Specifications (TS) affected by this report are listed below with the section and page for each one of the TS addressed in this COLR document.

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in the Introduction are presented below and listed sequentially by Technical Specification (TS). These limits have been developed using the NRC-approved methodologies specified in TS 6.9.1.7.

2.1 Reactor Core Safety Limits – Three Loops in Operation (TS 2.1.1)

- **Figure A1**(page 14B-A7) — In Modes 1 and 2, the combination of Thermal Power, reactor coolant system highest loop average temperature and pressurizer pressure shall not exceed the limits in Figure A1.

2.2 Reactor Trip System Instrumentation Setpoints (TS 2.2.1)

<u>NOTE 1 on TS Table 2.2-1 Overtemperature ΔT</u>

- **τ_1 = 0s, τ_2 = 0s** — Lead/Lag compensator on measured ΔT
- **τ_3 = 2s** — Lag compensator on measured ΔT
- **K_1 = 1.31**
- **K_2 = 0.023/°F**
- **τ_4 = 25s, τ_5 = 3s** — Time constants utilized in the lead-lag compensator for T_{avg}
- **τ_6 = 2s** — Lag compensator on measured T_{avg}
- **T′ ≤ 583.0 °F** — Indicated Loop T_{avg} at RATED THERMAL POWER
- **K_3 = 0.00116/psi**
- **P' ≥ 2235 psig** — Nominal RCS operating pressure
- **$f_1(\Delta I)$ = 0 for $q_t - q_b$ between – 18% and + 7%**.

  For each percent that the magnitude of $q_t - q_b$ exceeds – 18%, the ΔT Trip Setpoint shall be automatically reduced by 3.51% of its value at RATED THERMAL POWER; and

  For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 2.37% of its value at RATED THERMAL POWER.

  Where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

C31

NOTE 2 on TS Table 2.2-1 Overtemperature ΔT

The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.2% ΔT span for the Pressurizer Pressure channel, and 0.4% ΔTspan for the f(ΔI) channel. No separate Allowable Value is provided for Tavg because this function is part of the ΔT value.

NOTE 3 on TS Table 2.2-1 Overpower ΔT

C31

- **K_4 = 1.09**
- **$K_5 \geq$ 0.0/°F** For increasing average temperature
- **K_5 = 0.0/°F** For decreasing average temperature
- **$\tau_7 \geq$ 0 s** Time constants utilized in the lead-lag compensator for T_{avg}
- **K_6 = 0.0016/°F** For T > T”
- **K_6 = 0.0** For T ≤ T”
- **T” ≤ 583.0°F** Indicated Loop T_{avg} at RATED THERMAL POWER
- **f_2 (ΔI) = 0** For all ΔI

NOTE 4 on TS Table 2.2-1 Overpower ΔT

The Overpower ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel. No separate Allowable Value is provided for Tavg because this function is part of the ΔT value.

2.3 Shutdown Margin Limit for MODES 1, 2, 3 and 4 (TS 3.1.1.1)

- **Figure A2** (page 14B-A8)

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- **≥ 1.77 % Δk/k**

2.5 Moderator temperature coefficient (MTC) (TS 3.1.1.3)

- **≤ + 5.0 x 10^{-5} Δk/k/°F** — BOL, HZP, ARO and from HZP to 70% Rated Thermal Power (RTP)

- **From 70% RTP to 100% RTP the MTC decreasing linearly from ≤ + 5.0 x 10^{-5} Δk/k/°F to ≤ 0.0 x 10^{-5} Δk/k/°F**

- **Less negative than - 41.0 x 10^{-5} Δk/k/°F** — EOL, RTP, ARO

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- **Less negative than - 35.0 x 10^{-5} Δk/k/°F (-35 pcm/°F)** — Within 7 EFPD of reaching equilibrium boron concentration of 300 ppm.

The Revised Predicted near - EOL 300 ppm MTC shall be calculated using the algorithm contained in WCAP-13749-P-A:

Revised predicted MTC = Predicted MTC + AFD Correction - 3 pcm/°F

If the Revised Predicted MTC is less negative than the SR 4.1.1.3.b 300 ppm surveillance limit and all the benchmark criteria contained in the surveillance procedure are met, then an MTC measurement in accordance with SR 4.1.1.3.b is not required to be performed.

The neutronics methods used with WCAP-13749-P-A are those described in WCAP-16045-P-A, “Qualification of the Two-Dimensional Transport Code PARAGON,” August 2004. C31

2.7 Analog Rod Position Indication System (TS 3.1.3.2)

- **Figure A3** (page 14B-A9) — The All Rods Out (ARO) position for all shutdown Banks and Control Banks is defined to be 228 steps withdrawn. C31

2.8 Control Rod Insertion Limits (TS 3.1.3.6)

- **Figure A3** (page 14B-A9) — The control rod banks shall be limited in physical insertion as specified in Figure A3 for ARO = 228 steps withdrawn. C31

2.9 Axial Flux Difference (TS 3.2.1)

- **Figure A4** (page 14B-A10)

2.10 Heat Flux Hot Channel Factor $F_Q(Z)$ (TS 3.2.2)

- **$[F_Q]^L = 2.30$**

- **$K(z) = 1.0$** For $0' \leq z \leq 12'$ where z is core height in ft

2.11 Nuclear Enthalpy Rise Hot Channel Factor (TS 3.2.3)

- **$F_{\Delta H}^{RTP} = 1.600$ $PF_{\Delta H} = 0.3$**

2.12 DNB Parameters (TS 3.2.5)

- **RCS Tavg $\leq$ 585.0 °F**
- **Pressurizer Pressure $\geq$ 2204 psig**

Figure A1

Reactor Core Safety Limit – Three Loops in Operation

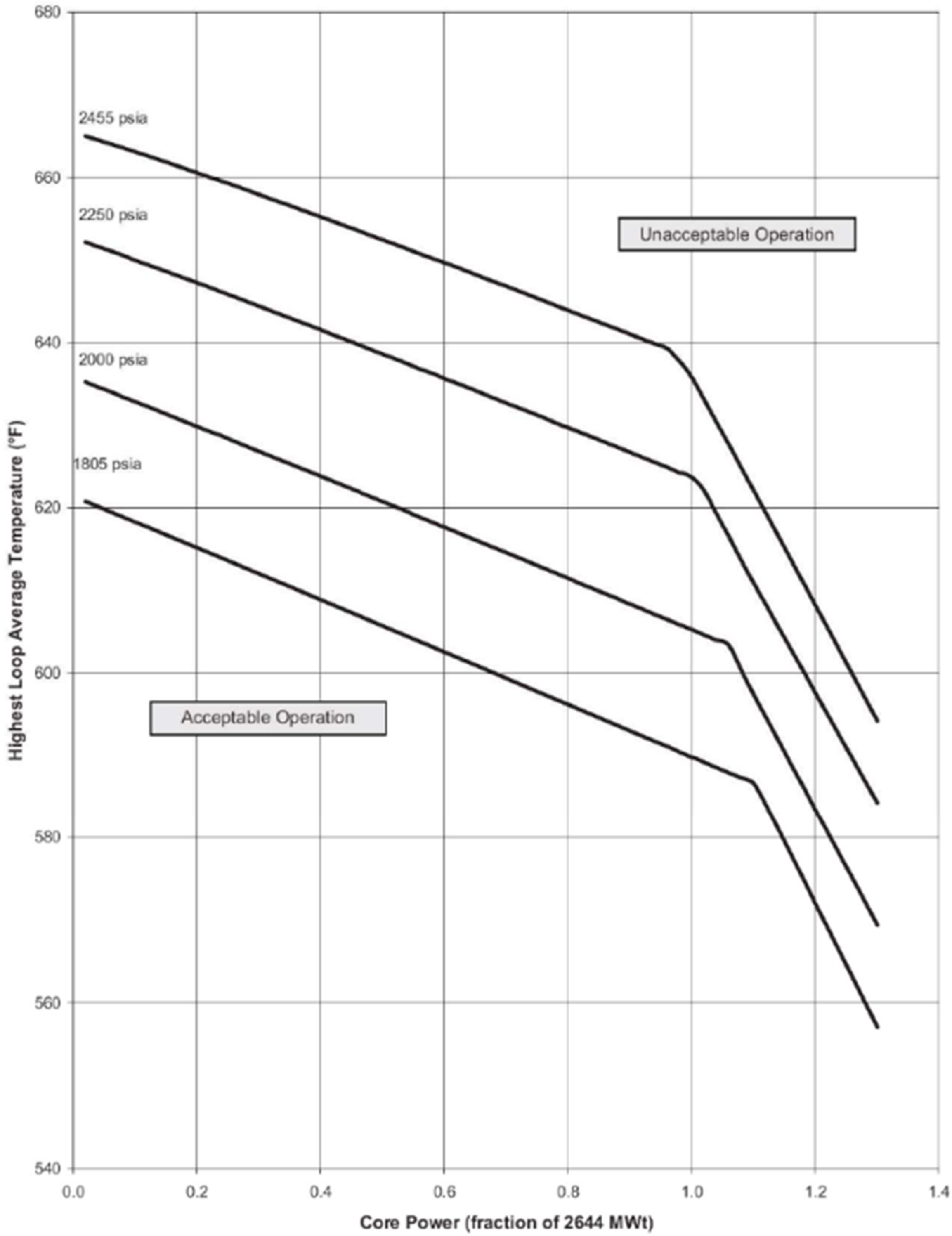

Figure A2

Required Shutdown Margin vs Reactor Coolant Boron Concentration

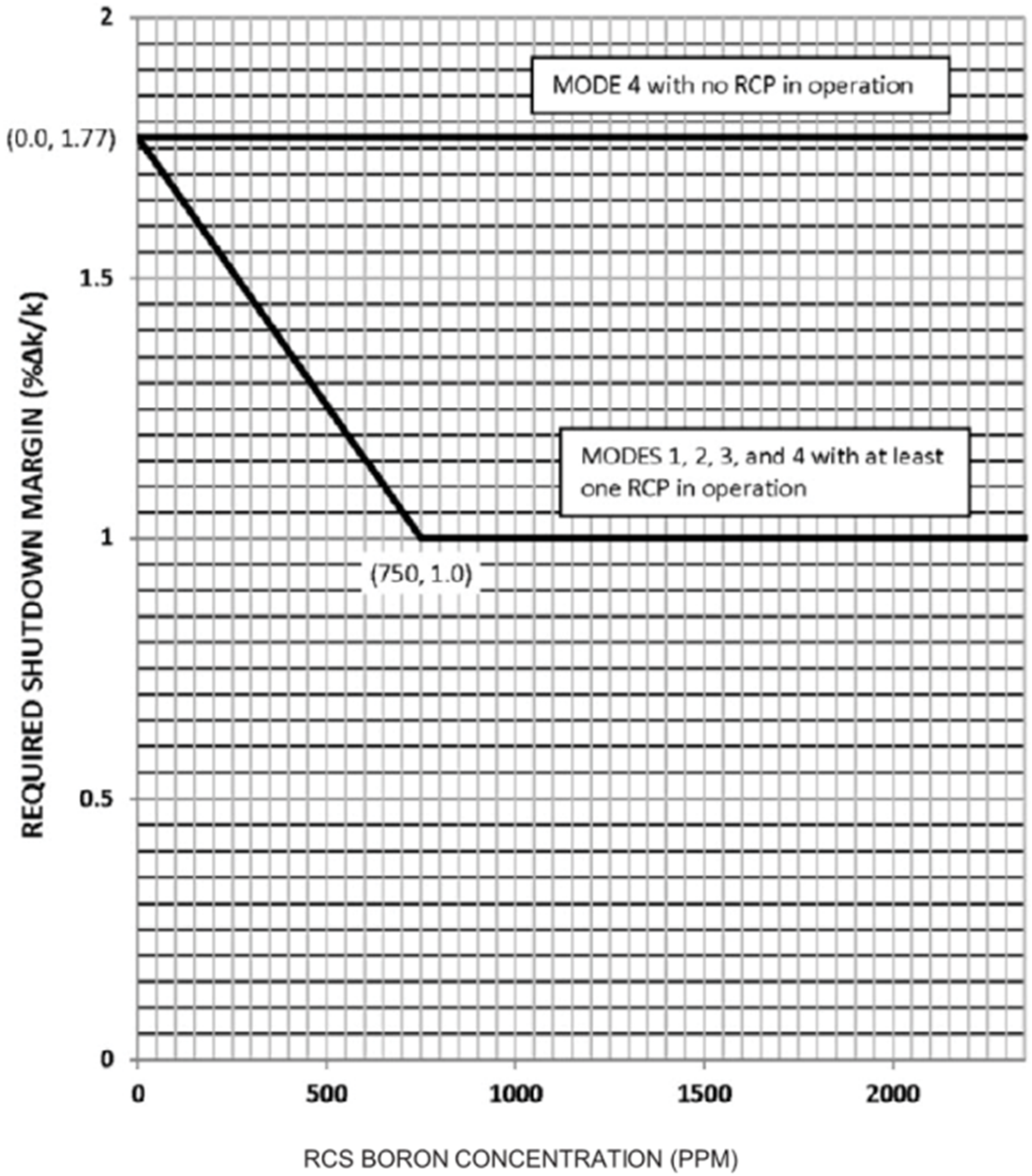

Figure A3

Turkey Point Unit 4 Cycle 32 Rod Insertion Limits vs Thermal Power
ARO = 228 Steps Withdrawn, Overlap = 100 Steps

C31

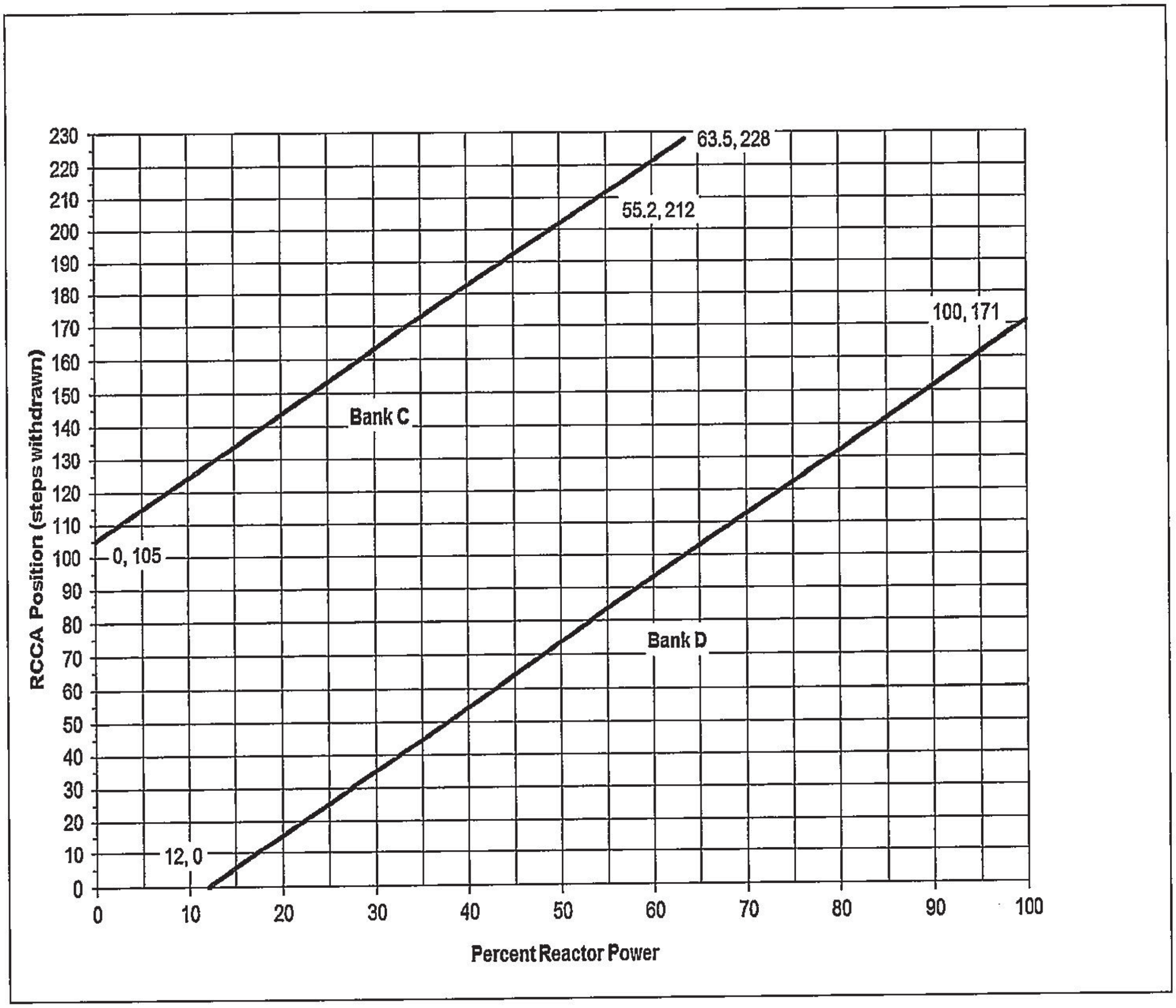

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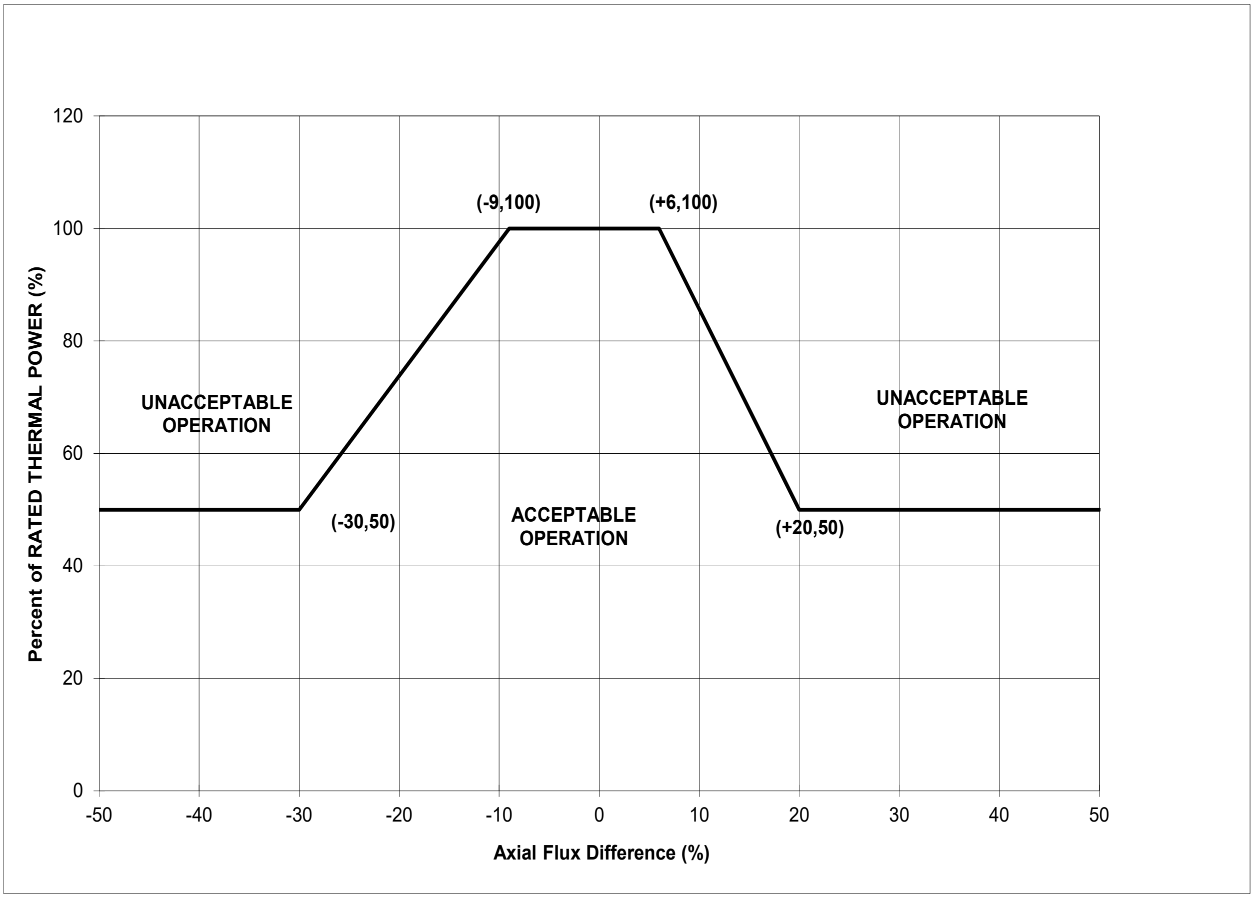
Figure A4

Axial Flux Difference as a Function of Rated Thermal Power
Turkey Point Unit 4 Cycle 32

C31



APPENDIX 14 C

TURKEY POINT UNITS 3 AND 4

UPDATED FSAR

MODIFICATION OF THE TURBINE RUNBACK SYSTEM

THIS APPENDIX HAS BEEN

ENTIRELY DELETED

SEE SECTION 10.2.2 “Turbine Runback (Load Cutback) Function” C29

FLORIDA POWER AND LIGHT COMPANY

APPENDIX 14D

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNITS 3 AND 4

DELETED IN ITS ENTIRETY

REFER TO CHAPTER 9, SECTION 9.5

AND CHAPTER 14, SECTION 14.2.1.3

HIGH DENSITY SPENT FUEL STORAGE RACKS

APPENDIX 14E

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNITS 3 AND 4

DELETED IN ITS ENTIRETY

REFER TO CHAPTER 9, SECTION 9.5

AND CHAPTER 14, SECTION 14.2.1.3

SPENT FUEL STORAGE FACILITY MODIFICATION

SAFETY ANALYSIS REPORT

APPENDIX 14F

ENVIRONMENTAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

This appendix contains the original licensing basis LOCA dose analysis. This analysis has been replaced with a revised analysis that can be found in Section 14.3.5.

The results of analyses described in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident (which has an exceedingly low probability of occurrence) are substantially less than the guidelines specified in 10 CFR 100. In summary, the computed thyroid dose values are (using the release assumption of TID-14844):

Integrated Dose	North Boundary Exclusion Radius 4164 ft	South Boundary 5582 ft	Low Population Distance 5 miles
0-2 hour dose, rem	93	65	9
0-31 day dose, rem	109	75	10

Loss-of-Coolant Accident

The loss-of-coolant accident has the potential for the highest off site doses, compared to all other accidents. The loss of coolant accident may result in a significant amount of clad rupture; however, since the fuel does not melt, only a limited quantity of fission products are released. If it is assumed that all the rods fail and that all the fission products in the gap spaces were released, the total release from the core would be less than 5% of the saturation quantities of the radioactive iodines and noble gases.

For analytical purposes the amount of radioactive fission products that could be released from the core have been calculated according to the fundamental assumptions given in Reference 1 (TID 14844). This calculational model has been widely used in evaluating the capability of PWR containment systems in the event of the core melt down. However, it should be pointed out that no accident of this magnitude has been described for these units; in fact, an accident of this magnitude is not considered credible.

The TID 14844 model assumes that 50% of the total core iodine inventory is released, and that one half of this amount becomes plated out onto surfaces within the containment. The remaining one half, or 25% of the total core iodine inventory, is assumed to be in the containment atmosphere and available for leakage. As a function of time the charcoal filter system collects and retains the iodine, and thereby the amount of iodine available for leakage is substantially reduced.

The TID 14844 model also assumes that 100% of the total core noble gas inventory and 1% of the total core solid fission product inventory are released into the containment.

Core Inventory of Iodines and Noble Gases

The total core inventory was calculated on the basis of the reactor having been operated as follows: (1) 2300 MW(t), (2) 625 days of full-power operation to produce 1-129 and the stable isotopes, and (3) except for I-129, full-power operation to reach the saturation inventory of the radioactive isotopes. Table 14F-1 gives information on the major iodine isotopes computed for the Turkey Point core, based on data given in TID 14844. Table 14F-2 gives information on the major noble gas isotopes.

Iodines and Noble Gases in Containment Atmosphere

The amount of noble gases in the containment atmosphere at time zero (according to the TID 14844 model) is the total amount listed in Table 14F-2. These gases are assumed to be completely mixed in the atmosphere, and available for leakage.

The amount of iodine in the containment atmosphere at time zero (according to the TID 14844 model, 25% of total) adds up to the following:

Total of I-127 and I-129	2,550 grams, stable
Total of I-131, I-132, I-133, I-134 and 1-135	152 grams, radioactive
Total Iodine in Containment	2,702 grams

The iodine, when released from the core, has been observed by those working in the field to be essentially composed of elemental iodine with little more than a trace of organic iodides. Upon reaching the containment, and as a function of time, some of the elemental iodine reacts with organic materials to form organic iodides, typified by methyl iodide. Also, some hydrogen iodide is formed.

The percentage of the iodine in the containment atmosphere that becomes converted into methyl iodide is not precisely known. The best evidence indicates that the value lies between an infinitesimal amount and 5%. It is stated in Reference 2 that "Although there is only a small amount of information available on which to base a judgement, a value of 10% for organic (nonremovable) iodides in the total available for leakage is considered very conservative...". For dose calculations the elemental iodine was taken as 95% and the methyl iodide as 5%.

With respect to iodine cleanup, the dose calculations are based on the removal that occurs only in the charcoal filter units and the 50% plateout previously mentioned. That is a conservative assumption since cleanup will also be achieved as follows:

1. Some iodine will be deposited on particles in the atmosphere. Some of these particles will be entrained by the containment borated spray water. The remainder of the particles will be collected in the HEPA filters.

2. Based on information given in Reference 3, and companion reports, the elemental iodine (and iodides other than organic) in the atmosphere may be effectively cleaned up by the containment spray water. This cleanup by the water is not permanent (since no iodine retaining agent is added) in that the iodine will seek an equilibrium distribution between the water and the air in accordance with its partition factor.

Iodine Cleanup With Emergency Containment Filter Units

The capability of the emergency containment filter units to collect elemental iodine and methyl iodide is indicated by a "decontamination factor" (DF), which in turn depends upon a "removal constant" (λ). Removal constants were computed on the basis of the equation and numerical values given in Table 14F-3. The following removal constants were computed:

Number of Filter Units Operating	Elemental Iodine λa	Methyl Iodide λb
3 (total installed)	3.53	2.74
2 (minimum safeguards)	2.35	1.83

The general decontamination factor equation is given in Table 14F-4. With the use of this equation the following decontamination factors were calculated, based on the iodine in the containment being composed of 0.95 elemental iodine and 0.05 methyl iodide:

Time period	2 Filter Units Operating DF	3 Filter Units Operating DF
0-2 hours	4.68	6.97
2-12 hours	> 100*	> 100*
12 hours - 31 days	> 100*	> 100*

Containment Assumptions

The containment design leak rate is 0.25% per day (2.9 x 10^{8}) fraction/sec) at the design pressure of 59 psig. In the event of a loss-of-coolant accident the containment pressure will rise to some value less than 59 psig, and will then decrease to near atmospheric pressure due to the action of the containment sprays and emergency containment coolers.

* These values were arbitrarily limited in order to obtain a finite number in the dose calculations.

For the dose calculations the pressure of the containment was assumed toremain at 59 psig for the entire length of the period, and thereby the leak rate was taken as a fixed value of 0.25% per day. This assumption tends to be
very conservative, particularly for the "12 hours-31 days" period.

Atmospheric Dispersion Model

For calculational purposes, the pressurized air-steam mixture in the containment was assumed to leak out at the established leak rate given above. This leakage from the containment becomes dispersed into the atmosphere and the dose rate to an individual at any specific location is a function of source concentration, time, distance, and atmospheric dispersion.

Dilution multipliers (x/Q), which reflect relative concentrations of radioactivity in the atmosphere as a function of distance from the containment, were calculated in accordance with equations and meteorological conditions given in Tables 14F-5 and 14F-6.

No credit was taken for the building wake effect for either the "2-12 hours" period or the "12 hours-31 days" period. This introduces some conservatism near the site boundary, but the error diminishes with distance. The values of σ_y and σ_z were taken from Reference 4.

The dilution multiplier values (in seconds/cubic meter) for the stated conditions at various locations are tabulated below:

Time period	North Boundary Exclusion Radius 4164 ft	South Boundary 5582 ft	Low Population Distance 5 miles
0-2 hours	154×10^{-6}	108×10^{-6}	15.0×10^{-6}
2-12 hours	108×10^{-6}	66×10^{-6}	6.5×10^{-6}
12 hours - 31 days	4.32×10^{-6}	2.64×10^{-6}	0.24×10^{-6}

Thyroid Dose Computations

The thyroid doses for various time periods were calculated according to the equation and values given in Table 14F-7.

The following values were obtained:

Integrated Dose	North Boundary Exclusion Radius 4164 ft	South Boundary 5582 ft	Low Population Distance 5 miles
0-2 hour dose, rem	93	65	9
0-31 day dose, rem	109	75	10

These values demonstrate that the amount of radioactivity that would be released to the environment in the event of a loss-of-coolant accident give dose values that are substantially less than the guidelines specified in 10 CFR 100.

Several parameter studies were performed in order to indicate the change in thyroid dose values that would result in the event of a deviation in an original assumption. For example, it was found that the doses remain almost unaffected in case of filter unit fan failure after a brief period of time.

The above given dose values were based on two filter units operating continuously for the duration of the accident. The principal cleanup occurs within the first two hours; in fact, within this period of time the iodine concentration will be reduced to less than 2% of the original concentration.

After two hours, the filter units serve to continue cleaning the air of residual amounts of iodine. The following tabulation illustrates the insensitivity of the dose values due to equipment malfunction after two hours.

Classification	Condition	Dose at Exclusion Radius, rem 0-2 hours	Dose at Exclusion Radius, rem 0-31 days
Normal	Two filter units operating 31 days or longer.	93	109
Abnormal	One filter unit operating 31 days or longer. Second filter unit operating for first 2 hours only.	93	110
Abnormal	Two filter units operating first 2 hours only.	93	111

In case a filter unit does fail after operating for a period of time, the radioactive decay heat is absorbed by the borated water spray system to the filters, thereby holding the collected iodine within the charcoal.

Another example is the sensitivity of the system to the methyl iodide content, since it cannot be established at this time precisely what fraction of the iodine will be in the methyl iodide form. Calculations were made to examine the variation in the 0-2 hour dose at the north boundary that would occur if the methyl iodide content in the containment atmosphere varied from 0% to as much as 15%.

	Methyl Iodide, Fraction .00	.05	.10	.15
DF	4.75	4.68	4.62	4.56
Dose, rem	92	93	94	95

For the calculations it was assumed that two filter units were operating with a λ of 2.35 for elemental iodine and a λ of 1.83 for methyl iodide, as given earlier. One concludes from the above that the exact amount of methyl iodide does not need to be known since the total dose varies very little.

A third example is the sensitivity of the system to unfilterable iodide. The concept of an unfilterable form of airborne iodine is hardly consistent with any physical model of filtration. It is possible, but not reasonable, on the basis of a thorough examination of the data (refer to references given in Reference 5), that some forms of iodine might be removed at very low efficiencies. It is a simplified approach to the calculations to assume that there is a form of iodine which is "unfilterable," or will be removed at zero percent efficiency, even though this does not agree with experimental data. In order to show sensitivity, calculations were made on the assumption of varying amounts of unfilterables to determine the variation in the 0-2 hour dose at the north and south boundaries, and the 0-31 day dose at a distance of 5 miles, with the unfilterable iodine varying in concentration from zero to 15% of the iodine concentration in the containment atmosphere. The results, with 2 filter units operating, were as follows:

	Fraction of Iodine that is Unfilterable:			
Integrated Dose	.00	.05	.10	.15
0-2 hr Dose, rem, North Boundary	92	108	125	142
0-2 hr Dose, rem, South Boundary	64	76	88	100
0-31 day Dose, rem, at 5 Miles	10	16	22	28

In reviewing the results computed on this basis, it is seen that the doses are all much less than 300 rem, even with the unfilterable content being 15%. Although the applicant does not believe that this calculational model is the proper one to use, it should be noted that the calculated dose values are low.

Short-term Thyroid Doses at Beach and Scout Camps

The maximum thyroid doses have also been considered for areas within the site boundary temporarily occupied by the public assuming the TID-14844 accident analysis model. These areas are the Turkey Point Beach at 2000 feet, the Girl Scout Camp at 2300 feet and the Boy Scout Camp at 2900 feet from the nearest containment structure. The respective X/Q values at these distances, considering the volume source correction, are 3.2×10^{-4}, 2.8×10^{-4} and 2.3×10^{-4} sec/M^3 for the period of 0 to 2 hours following the postulated LOCA.

By selection of a very conservative value of 59 psig maximum containment pressure for the leakage driving function over the entire initial two hours, the effective maximum containment leak rate is 0.25% / day. The resultant maximum two hour thyroid dose at the indicated locations, generated from an initial 95% elemental iodine and 5% methyl iodide atmospheric constituency, are:

Turkey Point Beach	190	rem
Girl Scout Camp	170	rem
Boy Scout Camp	138	rem

These values point out the requirement for the site evacuation procedure to be implemented within the initial 2 hour period, which will be provided and followed.

Whole Body Dose Computations

Whole body doses resulting from the accident were also computed. The major contribution is the dose from immersion in the plume. The direct radiation dose from the containment is insignificant due to the shielding provided by its walls.

Direct doses were calculated assuming immersion in a semi-infinite cloud containing a uniform distribution of the gas isotopes which have leaked from the containment. Cloud concentrations assumed were those actually calculated at the centerline of the plume.

The following whole body doses from the passing cloud were computed:

Integrated Dose	North Boundary Exclusion Radius 4164 ft	South Boundary 5582 ft	Low Population Distance 5 miles
0-2 hour dose, rem	3.1	2.2	0.4
0-31 day dose, rem	5.2	3.5	0.6

These values are small compared to the guidelines specified in 10 CFR 100.

Radiological Assessment of Containment Purge

The radiological doses due to a postulated loss of coolant accident presented in the proceeding analyses assumed that there was no containment purging occurring at the onset of the accident. Discussed herein are the results of an analysis performed to determine the incremental radiological dose at the site boundary and low population zone assuming the purge valves are fully open when the accident initiates and close upon receipt of signal as designed. These incremental doses, when added to those previously presented in Section 14.3.5, provide a maximum set of doses for a LOCA with containment purge. The results of this evaluation are presented in the following tables: (6)

THYROID DOSE (rem)

Location	LOCA	Increment Due To Purging	Total
Site Boundary - (0-2 hour)	93	10	103
Low Population Zone - (0-2 hour)	9	1	10

WHOLE BODY (rem)

Location	LOCA	Increment Due To Purging	Total
Site boundary - (0-2 hour)	3.1	.002	3.1
Low Population Zone - (0-2 hour)	.4	.0002	.4

The major assumptions which were used in the evaluation of the incremental dose are listed below:

1. The containment purge valves are closed 5 seconds after the containment high pressure signal is transmitted. There is a 2.7 second delay before

the increased containment pressure is detected which results in a total of 7.7 seconds for valve closure (8 seconds was conservatively assumed).

2. Radioactive releases via the purge valves during closure is from the Reactor Coolant System only.

3. The primary coolant iodine activity corresponds to the maximum limit of 30 μCi/gm Dose Equivalent.

4. It is conservatively assumed during the initial 8 seconds that 50% of the blowdown (worst FSAR case) from the break flashes and becomes homogeneously mixed in the containment atmosphere. All of the iodine in the flashed steam is assumed to become airborne.

5. The flow through the purge valves is assumed to be a mixture of steam and water. Frictionless flow through the valves is assumed.

6. FSAR meteorology is assumed.

7. Standard TID 14844 methodology was used to calculate the incremental doses.

The results clearly indicate that the anticipated dose caused by a LOCA with containment purging at the onset of the accident is well within the limits of 10 CFR 100.

References

1. J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, Calculation of Distance Factors for Power and Test Reactor Sites, USAEC Report TID-14844, March 23, 1961.

2. Supplemental Safety Evaluations by the Division of Reactor Licensing, United States Atomic Energy Commission, in the Matter of Florida Power and Light Company, Turkey Point Units 3 & 4, July 12, 1968.

3. Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1967, Oak Ridge National Laboratory, ORNL-4228, April 1968.

4. W. F. Hilsmeier and F. A. Gifford, Jr., Graphs for Estimating Atmospheric Dispersion, Report ORO-545, Weather Bureau Research Station, Oak Ridge, Tenn., August 23, 1962.

5. Supplement No. 14 to Application for Licenses, re Florida Power & Light Company, Turkey Point Units 3 & 4, USAEC Docket Nos. 50-250, 50-251, March 14, 1968.

6. R. E. Uhrig (FPL) letter #L-79-346, to A. Schwencer (NRC), dated December 13, 1979, "Containment Purge".

TABLE 14F-1

IODINE ISOTOPES AND THEIR ESTIMATED
QUANTITIES FOR A FULL CORE INVENTORY AT TIME ZERO

Isotope	Half-Life	Grams	Curies
I-127	Stable	2,040	0
I-129	1.72×10^7 years	8,170	~ 0
I-131	8.05 days	452	57.7×10^6
I-132	2.4 hours	8.25	87.5×10^6
I-133	20.8 hours	109.7	129.5×10^6
I-134	52.5 minutes	5.35	151.3×10^6
I-135	6.68 hours	31.9	117.2×10^6
Lumping all radioactive isotopes into an I-131 equivalent			$\approx 109 \times 10^6$

TABLE 14F-2

NOBLE GAS ISOTOPES AND THEIR ESTIMATED
QUANTITIES FOR A FULL CORE INVENTORY AT TIME ZERO

Isotope	Half-Life	Curies
Kr-83m	114 minutes	10.6×10^{6}
Kr-85	10.76 Years	0.83×10^{6}
Kr-85m	4.36 hours	25.5×10^{6}
Kr-87	78 minutes	47.3×10^{6}
Kr-88	2.77 hours	64.3×10^{6}
Xe-131m	12.0 days	0.46×10^{6}
Xe-133m	2.3 days	3.08×10^{6}
Xe-133	5.27 days	128.4×10^{6}
Xe-135m	15.6 minutes	41.5×10^{6}
Xe-135	9.13 hours	32.0×10^{6}

TABLE 14F-3
EQUATION FOR REMOVAL CONSTANT

$$\lambda = \frac{n\ v\ e\ m\ 60}{V}$$

λ = removal constant, per hour
n = number of filter units operating
v = atmosphere flow through each filter unit, cu ft/min
e = charcoal filter efficiency, fraction
m = atmosphere mixing factor, fraction
V = free volume of containment, cu ft

	Elemental iodine a	Methyl iodide b
v	37,500	37,500
e	0.9	0.7
m	0.9	0.9
v	1.55×10^6	1.55×10^6

TABLE 14F-4

GENERAL DECONTAMINATION FACTOR EQUATION

$$DF = \frac{1}{F_a e^{-\lambda_a t_1}\left(\frac{1-e^{-\lambda_a t_2}}{\lambda_a t_2}\right) + F_b e^{-\lambda_b t_1}\left(\frac{1-e^{-\lambda_b t_2}}{\lambda_b t_2}\right) + F_c}$$

F_a = filterable elemental iodine, fraction of total iodine in containment atmosphere.

F_b = filterable methyl iodide, fraction of total iodine in containment atmosphere.

F_c = unfilterable iodine and iodide; engineering tests indicate no components to be unfilterable; therefore, this is assumed to be zero.

t_1 = time of operation prior to the period under consideration, hours.

t_2 = time of operation during the period under consideration, hours.

TABLE 14F-5

DILUTION MULTIPLIER EQUATIONS

<u>Time period</u>

0-2 hours

$$\frac{\chi}{Q} = \frac{1}{\bar{\mu}\,(\pi\,\sigma_y\,\sigma_z + cA)}$$

2-12 hours

$$\frac{\chi}{Q} = \frac{1}{\beta\,\bar{\mu}\,\sqrt{\pi/2}\,\sigma_z\,x}$$

12 hours - 31 days

$$\frac{\chi}{Q} = \frac{f}{\beta}\sum\frac{F_i}{\bar{\mu}_i\,\sqrt{\pi/2}\,\sigma_{z_i}\,x}$$

X = concentration, curies/cu. meter
Q = source strength, curies/second
$\bar{\mu}$ = average wind speed, meters/second
$\bar{\mu}_i$ = wind speed for condition i, meters/second
σ_y= horizontal dispersion parameter, meters
σ_z= vertical dispersion parameter, meters
σ_{zi}= vertical dispersion parameter for condition i, meters
c = building shape factor (selected as 0.5)
A = cross-sectional area of building normal to wind (1750 sq meters)
β = sector size, radians
x = distance from source, meters
f = fraction of time wind blows in sector
F_i= fraction of time condition i exists

TABLE 14F-6

METEOROLOGICAL CONDITIONS

Time period	Condition
0-2 hours	Stability category, Pasquill F; Wind speed, 2 meters/sec; Wind direction, unvarying.
2-12 hours	Stability category, Pasquill F; Wind speed, 2 meters/sec; Wind direction,10 degree sector.
12 hours - 31 days	Wind direction, 22.5 degree sector; Wind blowing in this sector 25% of the time with the following variable conditions:

Fraction	Stability category	Wind speed meters/sec
.25	F	2
.50	D	5
.25	C	4

TABLE 14F-7

THYROID DOSE EQUATION AND SPECIFIC VALUES

$$\text{Dose (in rem)} = t\, B L \bar{A} \quad \frac{x}{Q} \quad \frac{1}{DF} \quad DCF$$

t = time period, hours

B = breathing rate, cu. meters/hour

L = reactor building leak rate, per second

$\bar{A}$ = average inventory of equivalent I-131 available for leakage assuming no filter unit cleanup during the period, curies

$\frac{x}{Q}$ = atmospheric dilution multiplier, seconds/cu. meter

DF = iodine decontamination factor for the period; that is, the ratio of iodine without cleanup to iodine with cleanup

DCF = dose conversion factor for I-131, rem/curie

	0-2 hours	2-12 hours	12 hours - 31 days
t	2	10	732
B	1.25	1.00	.834
L	2.9×10^{-8}	2.9×10^{-8}	2.9×10^{-8}
$\bar{A}$	26.17×10^{6}	23.04×10^{6}	5.24×10^{6}
DF (2 units)	4.68	100	100
DCF	1.48×10^{6}	1.48×10^{6}	1.48×10^{6}

$\frac{x}{Q}$ Refer to tabulation given in paragraph "Atmospheric Dispersion Model".

APPENDIX 14G

HISTORICAL DISCUSSION OF CONTAINMENT PRESSURE TRANSIENT MARGINS ASSOCIATED WITH CONTAINMENT STRUCTURAL PRESSURE OF 59 PSIG

INTRODUCTION

This appendix contains the original FSAR discussion of the containment design pressure margins associated with the original containment structural capability pressure of 59 psig. Since the original containment structural capability pressure of 59 psig has been replaced with the licensed design basis pressure (55 psig) approved by the Atomic Energy Commission (AEC) during the operating license stage, this discussion is of historical importance only and does not apply to the current licensed containment design pressure or to the basis for calculating the minimum required prestress forces for the containment post-tensioning system. Refer to the engineering evaluation contained in Reference 1.

BACKGROUND

The licensed containment design basis pressure of 55 psig was established during the very early stages of plant licensing and has carried through to current licensing documents. The PSAR and FSAR indicated that a 55 psig reference containment design pressure was conservatively established for the design basis (29-inch double-ended pipe break) loss-of-coolant accident (LOCA), based on a 49.9 psig calculated peak pressure plus a 10% safety margin; and the structural proof test was conducted at 115% design pressure to check structural integrity. Refer to PSAR Sections 5.4.1.a and 12.2.3 (Reference 2), and to original (1970) FSAR Section 5.1.1, (Reference 3).

Other LOCA study cases, assuming partial safeguards availability, were also considered. These study cases did not constitute licensed design basis accident scenarios, but rather provided an indication of potential containment performance requirements beyond-the-licensing-basis for purposes of establishing conservative design margins for the containment structures.

These scenarios were developed in response to Atomic Energy Commission (AEC) questions, and to address uncertainties as to the availability of primary system accumulators. As a result, some of these other cases assumed partial safeguards operation with no core cooling, which were conditions that are beyond the required postulation of a single active or passive failure. Refer to PSAR Supplement 2, Questions 1.0 and 3.0 (Reference 4). For instance, the AEC requested that a "no-core-cooling" case be considered, in which partial safeguards equipment, operating on diesel power, introduced all the safety injection water directly into the sump. This case resulted in a maximum pressure of 58.5 psig. However, the value of 55 psig came about as the result of the design basis analysis which assumed that partial safeguards equipment, operating on diesel power, provided core cooling by having 2/3 of the safety injection water flow paths reach the core.

To accommodate these hypothetical, beyond-the-licensing-basis scenarios, the containment structure was designed with additional margins to withstand a pressure of 59 psig; however, the licensed design basis LOCA analysis calculated peak pressure was 49.9 psig, and "55 psig [was] considered as nominal structural design pressure, thus allowing a margin of 10% over the calculated peak accident pressure." Refer to original 1970 FSAR, Section 5.1.1 - Reference 3).

CONTAINMENT MARGIN EVALUATIONS

Evaluation of the capability of the containment and associated cooling systems to absorb energy additions without exceeding the containment design pressure requires consideration of two periods of time following a postulated large area rupture of the reactor coolant system.

The first period is the blowdown phase. Since blowdown occurs too rapidly for the containment cooling systems to be activated, there must be sufficient energy absorption capability in the free volume of the containment (with due credit for energy absorption in the containment structures) to limit the resulting pressure below design.

The second period is the post-blowdown period where the containment cooling systems must be able to absorb any postulated post-blowdown energy additions and continue to limit the containment pressure below design.

Margin - Blowdown Peak to Design Pressure

Point A in Figure 14G-1 corresponds to the internal energy at the end of a DE break blowdown, 195 x 10^6 Btu. In order for the pressure to increase to design pressure (59 psig) the internal energy must be increased to 231 x 10^6 Btu (Point B). The allowed energy addition is therefore 36 x 10^6 Btu. Since energy transferred to the containment from the core is in the form of steam the total transferred core energy corresponding to allowed energy addition is as follows:

$$Q_{core} = \frac{h_{fg}}{h_g} Q_{allowed} = 36 \times 10^6 \times \frac{921.9}{1177.6} = 28.4 \times 10^6 \text{ Btu}$$

This allowable value of energy which could be transferred from the core to the containment without increasing the transient containment pressure to design pressure can be compared to the energy stored in the reactor vessel and transferred to the steam generator during blowdown for the double ended break. The thick metal of the reactor vessel was not considered since a negligible amount of this energy can be transferred in the short blowdown time.

Stored in the core	15.0 x 10^6 Btu
Core internals Metal	0.3 x 10^6 Btu
Transferred to Steam Generators	1.4 x 10^6 Btu
	16.7 x 10^6 Btu

Thus, the containment has the capability to limit containment pressure below design even if all of the available energy sources were transferred to the containment at the end of blowdown. This would also include no credit for energy absorption in the steam generator. For this to occur an extremely high core to coolant heat transfer coefficient is necessary. This would result in the core and internals being completely subcooled and limit the potential for release of fission products.

Additional Energy Added as Superheat

Line A to C on Figure 14G-1 represents a constant mass line extended into the superheated region. Comparison of the energy addition allowable for the superheated case relative to the saturated case shows a lesser ability of the containment to absorb an equivalent amount of energy as superheat. An addition of 8.5 x 10^6 Btu of energy after blowdown would cause the containment pressure to increase to design. The recombination of hydrogen and oxygen from 9.6% Zr-H_2O reaction completed before the end of blowdown would be required to generate 8.5 x 10^6 Btu's of energy. For the case analyzed, the core was assumed to be in a subcooled state, and no Zr-H_2O reaction would be possible. In order for Zr-H_2O reaction to occur before the end of blowdown all of the stored initial energy must remain in the core. If this occurred a blowdown peak containment pressure of only 44.2 psig would be reached instead of 49.9 psig in the case analyzed. Lines D and E on Figure 14G-1 represent the superheat energy addition required to increase the pressure to the design pressure and this corresponds to the hydrogen oxygen recombination energy from a 15.8% Zr-H_2O reaction.

It is, therefore, concluded that the containment has the capability to absorb the maximum energy addition from any loss-of-coolant accident without reliance on the containment cooling system. In addition, a substantial margin exists for energy additions from arbitrary energy sources much greater than any possible.

Margin - Post Blowdown Energy Additions

The Safety Injection System is designed to rapidly cool the core and stop significant addition of mass and energy to the containment.

However, the following cases are presented to demonstrate the capability of the containment to withstand post accident energy additions without credit for core cooling.

Case 1 : Blowdown from a large area rupture with continued addition of the core residual energy and hot metal energy to the containment as steam.

Case 2 : Same as Case I but with the energy addition from a maximum Zirconium - water reaction.

Figure 14G-2 presents the containment pressure transient for Case 1. For this case the decay heat generated for a 2300 MWt core operated for an infinite time is conservatively assumed. This decay heat is added to the containment in the form of steam by the boiling off of water in the reactor vessel. For this case injection water merely serves as a mechanism to transfer the residual energy to the containment as it is produced. Injection water is in effect throttled at the required rate.

In addition, all the stored energy in the core and internals which is calculated to remain at the end of blow down is added in the same way during the time interval between 12.7 and 36.5 seconds (corresponds to accumulator injection time). Also all the sensible heat of the reactor vessel is added as steam exponentially over 2000 seconds time interval.

The containment cooling system capability assumed in the analysis was one of two available containment spray pumps and two of three available emergency containment coolers. This is the minimum equipment available considering the single failure criterion in the emergency power system, the containment spray system and the fan cooler system.

The containment heat removal capability started at 60 seconds exceeds the energy addition rate and the pressure does not exceed the initial blowdown value. An extended depressurization time results due to the increased heat load on the containment coolers.

It should be emphasized that this situation is highly unrealistic in that continued addition of steam to the containment after blowdown could not occur. The accumulator and Safety Injection System acts to rapidly reflood and cool the core.

Figure 14G-3 presents the containment pressure transient for Case 2. To realistically account for the energy necessary to cause a metal-water reaction, sufficient energy must be stored in the core. Storing the energy in the core rather than transferring it to the coolant causes a decrease in the blowdown peak.

The reaction was calculated using the parabolic rate equation developed by Baker and assuming that the clad continues to react until zirconium oxide melting temperature of 4800°F is reached. An additional 10% reaction of the unreacted clad is assumed when the oxide melting temperature is reached. A total reaction of 32.3% has occurred after 1000 seconds. If the reactions were to be steam limited, they could result in a higher total reaction but at a much later time. The reaction provided by the parabolic rate equation therefore, imposes the greatest load on the containment cooling system.

As in Case 2, the residual heat and sensible heat is added to the containment as steam. The energy from the $Zr-H_2O$ reaction is added to the containment as it is produced. The hydrogen was assumed to burn as it entered the containment from the break.

The blowdown peak was reduced to 44 psig and a peak pressure of 57.7 psig was reached at 600 seconds. At this time the heat removal capability of the containment cooling system assumed to be operating (one containment spray pump and two fan coolers) exceeded the energy addition from all sources.

For comparison the containment pressure transients for Cases 1, 2 and the double ended blowdown are replotted in Figure 14G-4. It is concluded that operation of the minimum containment cooling system equipment provides the capability of limiting the containment pressure below its design pressure with the addition of all available energy sources and without credit for the cooling effect from the safety injection system.

DISCUSSION OF ENERGY SOURCES USED IN CASES 1 AND 2

The following is a summary of the energy sources and the containment heat removal capacities used in the containment capability study. Figure 14G-5 presents the rate of energy addition from core decay heat, Zr-H_2O reaction energy, and the hydrogen-oxygen recombination energy. The heat removal capability for the partial containment cooling (one spray pump and two fan coolers) is also presented. These heat removal values are for operation with the containment at design pressure.

The integrated heat additions and heat removals for Cases 1 and 2 are plotted in Figures 14G-6 and 14G-7, respectively. These curves are presented in a manner that demonstrates the capability of the containment and the cooling systems to absorb energy. The integrated heat removal capacity is started at the internal energy corresponding to design pressure, while the integrated heat additions begin from the internal energy calculated at the end of blowdown for each case. The upper line on each curve is the containment structures and containment cooling systems capability to absorb energy additions without exceeding design pressure. The lower curve for each are the energy addition curves, and since these energy additions are the maximum possible with no credit for core cooling, there is more than adequate capability to absorb arbitrary additions.

The curves in Figures 14G-8 and 14G-9 present the individual contribution of the heat removal and heat addition source, respectively.

REFERENCES

1. Engineering Evaluation JPN-PTN-SENP-93-008,"No Significant Hazards Evaluation Related to Containment Design Pressure Technical Specification and UFSAR Changes," Revision 0, dated April 23, 1993.

2. Turkey Point Units 3 and 4 Preliminary Safety Analysis Report (PSAR), Sections 5.4.1.a and 12.2.3, submitted by Application dated March 22, 1966.

3. Turkey Point Units 3 and 4 (original) Final Safety Analysis Report (FSAR), Section 5.1.1, "Containment Structure Design Bases," Revision 4, dated August 12, 1970.