Florida Power & Light, 9760 S.W. 344 St. Homestead, FL 33035

FPL

JUL 22 2011
L-2011-237
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
Supplemental Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Containment and Ventilation Issues

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 – Containment and Ventilation (SCVB) Request for Additional Information - Round 1," Accession No. ML110950084, April 1, 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 April 1, 2011 [Reference 2], additional information regarding containment analysis and ventilation issues was requested by the NRC staff in the Containment and Ventilation Branch (SCVB) to support their review of the EPU License Amendment Request (LAR). The RAI consisted of eleven (11) questions regarding control room heating and ventilation, main feedwater system modifications, combustible gas control in containment, generic letter (GL) 96-06 over-pressurization concerns for piping between containment isolation valves, containment design basis accident analyses, and emergency core cooling system (ECCS) performance required for their review of the EPU LAR. On April 28, 2011, FPL provided its responses to these RAI questions via FPL letter L-2011-084 [Reference 3]. In addition, FPL indicated that a supplemental response would be provided by July 29, 2011 for RAI question SCVB-1.10 regarding the impact of the identified EPITOME code errors on the Loss-of-Coolant-Accident (LOCA) containment response analysis. The supplemental response was to include a revised LOCA containment response analysis and address all other areas impacted. Accordingly, RAI question SCVB-1.10 and FPL's supplemental response with the revised containment response analyses and assessment of all affected areas as well as proposed changes to TS 3.4.6.1, 3.6.1.4, and Table 3.3-4 are documented in the Attachment to this letter.

A001
A002
-NRR

The Turkey Point Plant Nuclear Safety Committee has reviewed the proposed changes to the Technical Specifications and has concluded that the changes do not involve a significant hazards consideration. This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1].

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

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

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 July 22, 2011.

Very truly yours,

Eric McCartney for M. Kiley

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachment

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 SCVB CONTAINMENT AND VENTILATION ISSUES

ATTACHMENT

Response to Request for Additional Information

The following information is provided by Florida Power and Light Company (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) No. 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL letter L-2010-113 on October 21, 2010 [Reference 1].

On April 1, 2011, FPL received an RAI via email from the NRC Project Manager (PM) regarding containment analysis and ventilation issues associated with FPL's request to implement the Extended Power Uprate [Reference 2]. The RAI consisted of eleven (11) questions from the NRC Containment and Ventilation Branch (SCVB) regarding control room heating and ventilation, station blackout heating and air conditioning requirements, main feedwater system modifications, combustible gas control in containment, generic letter (GL) 96-06 over-pressurization concerns for piping between containment isolation valves, containment design basis accident analyses, and emergency core cooling system (ECCS) performance required for their review of the EPU LAR. On April 28, 2011, FPL provided its responses to these RAI questions via FPL letter L-2011-084 [Reference 3]. In addition, FPL indicated that a supplemental response would be provided by July 29, 2011 for RAI question SCVB-1.10 regarding the impact of the identified EPITOME code errors on the Loss-of-Coolant-Accident (LOCA) containment response analysis. The supplemental response would include a revised LOCA containment response analysis and address all other areas impacted. Accordingly, RAI question SCVB-1.10 and FPL's supplemental response with the revised containment response analyses and assessment of all affected areas are documented below.

Containment and Ventilation

SCVB-1.10

Section 2.6.3.1, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accident," of Attachment 4 to the LAR, subsection 2.6.3.1.2.1.3 discusses the evaluation model used for the long-term LOCA M&E release calculations. Westinghouse discovered that the computer code (EPITOME) used to generate the M&E inputs for the containment peak pressure analysis contains an error which could result in an increase in the containment pressure and temperature for the double ended pump suction LOCA, including a maximum increase in the peak containment pressure of up to 5 psi and temperature of up to 5.5 deg F. Since FPL used the EPITOME code in its EPU analysis for Turkey Point, provide information on how this code error affects Turkey Point's containment response analysis for a LOCA, specifically regarding the calculated peak pressure and peak temperature.

Summary of EPITOME Issue

Westinghouse has identified six potential issues associated with another licensee's LOCA mass and energy (M&E) analysis. Only one of these issues, the EPITOME computer code, is applicable to the Turkey Point LOCA M&E release analysis originally submitted for the EPU. Westinghouse identified this issue as a result of an inconsistency in calculations performed by the EPITOME computer code that impacts post blowdown LOCA M&E calculations. This error only impacts the double ended pump suction (DEPS) break; the double ended hot leg break (DEHL) is not affected by the EPITOME issue. As a result of this issue, the EPITOME computer code has been revised, and the modifications made to the code do not invalidate the Safety Evaluation Report (SER) for WCAP-10325-P-A [Reference 4].

The LOCA M&E release analysis and associated containment integrity analysis were revised to address the specific issue identified above. The revised LOCA containment integrity analysis results compared to the analyses and evaluations documented in Reference 1 are provided below:

1. Peak LOCA containment vapor temperature increased by 3.0°F to 281.8°F (See Figure 2).
2. Peak LOCA containment pressure increased by 0.5 psi to 53.9 psig (See Figure 1). The design limit is 55 psig.
3. Peak LOCA structural temperature increased by 3.9°F to 273.5°F. The design limit is 283°F.
4. Peak LOCA containment sump temperature at initiation of recirculation decreased by 7.0°F to 237.8°F.
5. Maximum component cooling water (CCW) heat exchanger outlet temperature (supply) increased by 0.8°F to 158.6°F.
6. Maximum residual heat removal (RHR) heat exchanger outlet temperature increased by 2.0°F to 200.8°F.
7. Maximum RHR heat exchanger CCW outlet temperature increased by 2.0°F to 193.0°F.
8. Maximum emergency containment cooler (ECC) cooling water outlet temperature increased by 10.3°F to 205.4°F.
9. Maximum CCW heat exchanger inlet temperature (return) increased by 2.9°F to 185.4°F.

The steamline break (SLB) M&E release and containment response analyses do not utilize the EPITOME code and were not directly impacted by the issues discussed. The SLB analyses were revised to address some of the other input considerations described below. The cases necessary to support the complete spectrum of SLB containment response cases from the EPU analysis were reanalyzed. The revised SLB containment integrity analysis results as compared to the analyses used as input to Reference 1 are provided below:

1. Peak SLB containment temperature increased by 14.5°F to 317.0°F (See Figure 4).
2. Peak SLB containment pressure increased by 1.1 psi to 53.4 psig (See Figure 3). The design limit is 55 psig.
3. Peak SLB structural temperature increased by 15.3°F to 279.2°F. The design limit is 283°F.
4. Maximum SLB component cooling water (CCW) heat exchanger inlet temperature (return) increased by 4.4°F to 179.7°F.
5. Maximum SLB CCW heat exchanger outlet temperature (supply) increased by 2.9°F to 151.8°F.
6. Maximum SLB emergency containment cooler (ECC) cooling water outlet temperature increased by 5.8 F to 202.1°F.

Other Input Considerations (Containment Volume, other changes, etc.)

In addition to reanalyzing the LOCA containment analysis to correct the EPITOME error, it was discovered that the values assumed in the Turkey Point EPU analysis for the containment volume, the initial containment temperature, and the hot leg recirculation injection flows required revision. Some of the changes discussed below also impacted the SLB analysis. All changes made comply with the Westinghouse NRC-approved methodology.

Containment Volume

The analysis value for the containment free volume was changed from the previous single value (1.55E6 ft^3), to a range of values (1.45E6 ft^3 to 1.60E6 ft^3). The LOCA and SLB containment analyses are performed using a minimum value for the free volume, so the re-analyses were performed with the volume set at 1.45E6 ft^3.

Initial Containment Temperature

The analysis value for the initial containment temperature increased from 126 °F to 130 °F. The LOCA and SLB analyses were performed with the initial containment air temperature, the initial temperature of the heat sinks, and the initial temperature of the fluid contained in the accumulators to support 130°F. The Technical Specifications define the maximum operating temperature to be 120°F with an allowance up to but not to exceed 125°F for a specified period of time. This value bounds the current Technical Specification value.

Hot Leg Recirculation Injection Flow

The hot leg recirculation flow to the reactor coolant system (RCS) supporting the LOCA long-term equipment qualification (EQ) case was modified as a result of a hydraulic calculation correction.

The EPITOME error has an adverse effect on the LOCA M&Es during the reflood phase. The decreased analysis value for containment volume, the increased initial containment temperature, and hot leg recirculation flow change all have adverse effects on the calculated containment response. To accommodate these adverse effects, the following additional changes have been made in the LOCA analysis. The corresponding changes were also incorporated into the SLB analysis when applicable.

Revised Initial Containment Conditions

The following changes are made to the containment initial conditions for the LOCA and SLB analyses. The initial containment pressure value is reduced compared to the previously assumed input. This input is defined within the Technical Specifications and will require a TS change (attached). With respect to relative humidity, the revised value has been justified based upon a plant-specific calculation.

Input Parameter	**Previous Value (Ref 1)**	**New Value**
Initial Containment Pressure (psia)	17.7	16.1
Initial Containment Relative Humidity (%)	5	30

Revised Decay Heat Model

A Turkey Point plant-specific decay heat curve has been generated for use in the present LOCA analysis. The blowdown phase M&E releases continue to be based on the generic Westinghouse decay heat curve. The reflood phase releases and the long-term decay heat are based on the Turkey Point plant-specific decay heat curve. The curve is slightly reduced from the previously used curve [Reference 1], for times less than approximately 10,000 seconds. The applicability of the Turkey Point plant-specific decay heat curve is confirmed each cycle as part of the normal reload safety analysis checklist (RSAC) process.

Containment Spray Droplet Size

The LOCA and SLB analysis value for the containment spray droplet size was reduced from 1,000 microns to 700 microns. The Turkey Point droplet size is defined by the plant-specific spray nozzle hardware.

Emergency Containment Cooler Operation

The control scheme for the emergency containment coolers (ECCs) will be modified. If one of the two normally actuated ECCs fails to start (e.g., due to failure of one diesel generator), the third (swing) ECC will be automatically loaded onto the energized bus. This ensures that exactly two ECCs will operate automatically without the possibility of three ECCs operating during accident conditions. The LOCA containment analysis value for the swing ECC start time is one minute following the event initiation. This design change requires a plant modification that restores an original design feature. Per TS 3.6.2.2, three ECCs are currently required to be operable and by crediting two of the ECCs, the containment analysis satisfies the single failure criteria.

Containment Spray Flow Rate

The containment spray flow rate is calculated as a function of the prevailing containment pressure. The Reference 1 calculation (for the LOCA injection phase) assumes a constant (low) refueling water storage tank (RWST) level for both phases of the RWST draindown since this minimizes the spray pump suction pressure and, therefore, minimizes the delivered spray flow rate. During the first phase of the RWST draindown, the spray flows calculated at a higher, yet applicable RWST level were assumed in the revised analysis. For the second phase of the RWST draindown, the low RWST level was assumed in the spray flow calculation. This change remains conservative but provides a slight increase in the delivered spray flow rate early in the LOCA event. The SLB flowrates were originally calculated based on the low RWST level and were not revised.

RCS Temperature and Pressure

The maximum operating average RCS temperature at full power is reduced to 581.5 °F consistent with planned operation limits. The maximum full power average RCS temperature assumed in the revised LOCA analysis is 587.2°F including an allowance for the final calculated instrument uncertainty. The lower operating temperature of 587.2°F is based upon the reduced temperature band and the final calculated uncertainty. The analysis value for the instrument uncertainty on pressurizer pressure is reduced to 37.7 psi, based on the final calculated uncertainty resulting in an analyzed pressure of 2287.7 psia.

Steam Generator Secondary Inventory

The initial steam generator secondary mass was revised for the LOCA analysis based upon the change in the full-power RCS temperature, reduction in uncertainties discussed above, and the removal of an additional allowance for steam generator level uncertainty in the calculation of the assumption for initial steam generator mass. Per WCAP-10325 methodology [Reference 4], the inclusion of steam generator level uncertainty is not required, but the modeled initial secondary mass is maximized from the calculated steam generator mass by a 10% increase.

Summary of LOCA Results

a) Containment Integrity

The LOCA containment integrity analysis was revised to address the applicable issues associated with the LOCA mass and energy release analysis and corresponding containment analysis identified by both Westinghouse and FPL, and the revised LOCA containment results were evaluated against the previous EPU evaluations submitted in Reference 1. The reanalysis resulted in an increase of 0.5 psi with a peak of 53.9 psig. The revised results meet the design limit of 55 psig. The same case calculates an increase of 3.9°F to 273.5°F on the peak structural temperature which meets the design limit of 283°F. Therefore, the proposed EPU remains acceptable with the revised EPU LOCA containment analysis.

b) Pressure/Temperature Curves

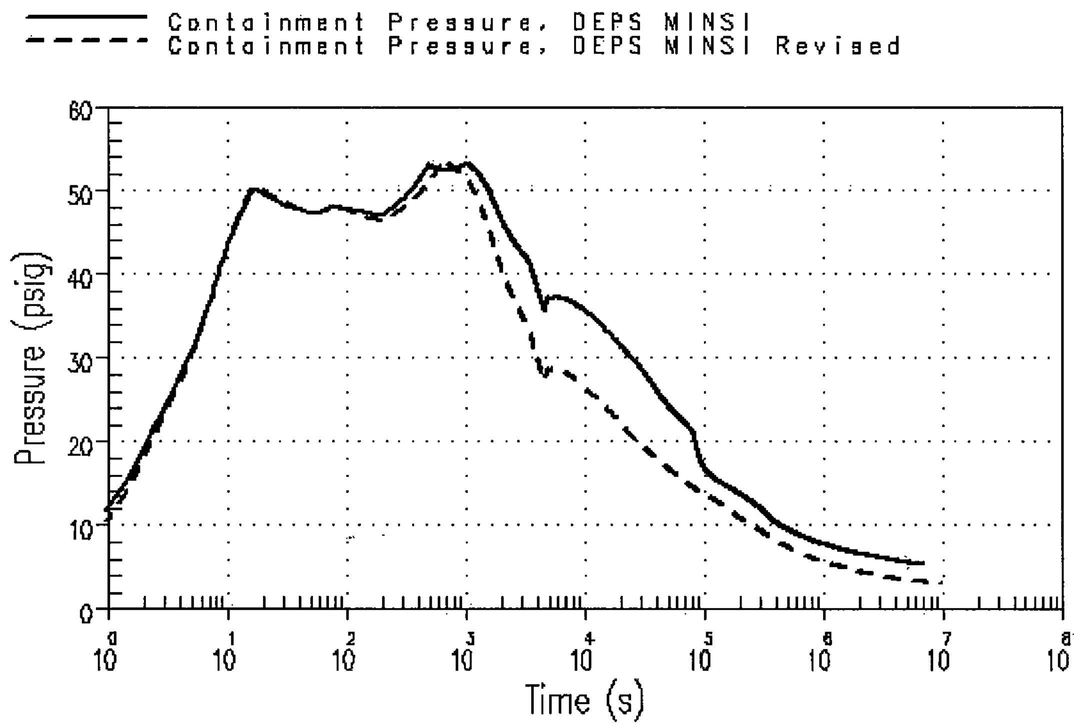

Figure 1: Turkey Point Units 3 and 4 Containment Pressure Comparison for DEPS MINSI (Double Ended Pump Suction, Minimum Safeguards) Case

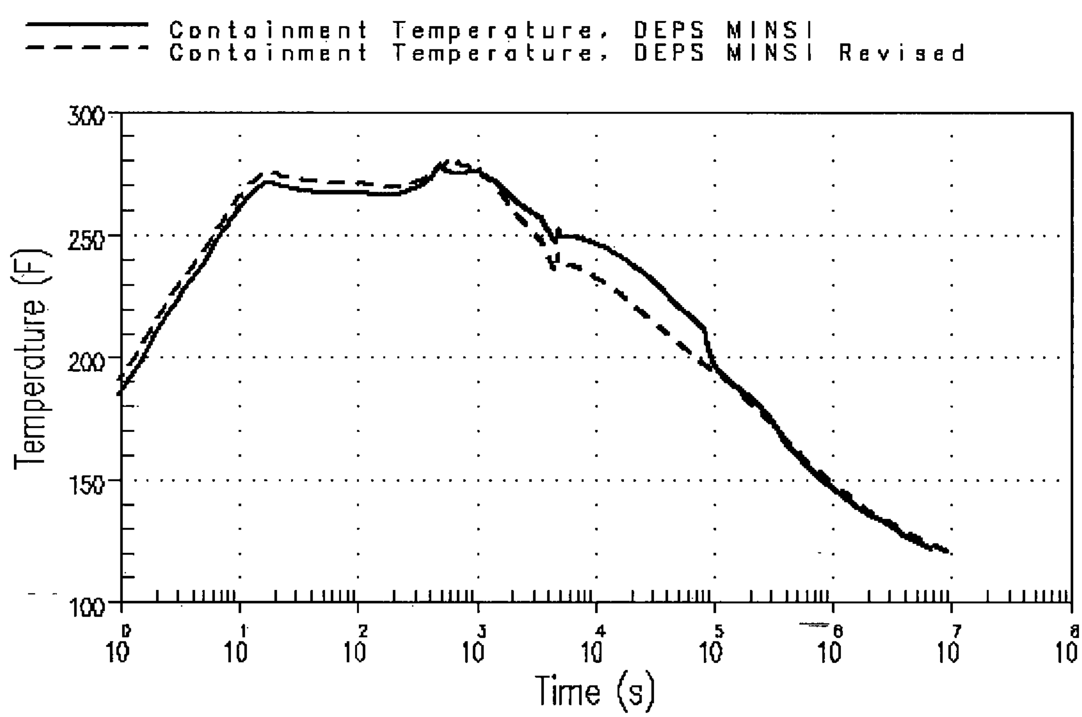

Figure 2: Turkey Point Units 3 and 4 Containment Temperature Comparison for DEPS MINSI (Double Ended Pump Suction, Minimum Safeguards) Case

c) Mass and Energy Tables

Table 1: DEPS Break Blowdown M&E Release

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
0.000	0.0	0.0	0.0	0.0
0.001	82893.5	45363.9	40200.6	21957.5
0.002	41410.8	22619.3	41056.1	22423.4
0.1	40943.0	22450.9	20320.5	11086.0
0.2	41741.4	23079.1	23081.4	12606.7
0.3	42703.6	23869.4	24455.5	13365.7
0.4	43677.4	24724.9	24343.8	13310.7
0.5	44333.7	25436.2	23678.1	12953.4
0.6	44336.0	25760.6	23108.0	12649.1
0.7	43367.0	25482.1	22836.6	12508.1
0.8	41959.8	24907.3	22741.5	12461.7
0.9	40691.5	24383.3	22636.0	12408.1
1.0	39601.9	23948.7	22495.2	12333.8
1.1	38585.4	23545.2	22329.9	12245.1
1.2	37590.6	23144.8	22156.3	12151.1
1.3	36676.9	22772.2	21983.5	12056.8
1.4	35852.7	22437.2	21809.8	11961.5
1.5	35093.5	22137.4	21645.7	11871.1
1.6	34329.3	21847.6	21498.6	11790.1
1.7	33519.2	21549.1	21362.2	11714.8
1.8	32440.3	21098.5	21211.7	11631.6
1.9	31207.5	20551.2	21033.0	11532.4
2.0	29658.3	19779.7	20650.6	11320.5
2.1	27953.3	18865.8	20153.8	11045.9
2.2	25863.1	17624.3	19710.6	10801.5
2.3	23653.7	16210.1	19271.2	10559.4
2.4	21388.8	14689.0	18789.4	10294.5
2.5	19977.1	13741.4	18384.3	10072.2
2.6	19829.0	13660.2	18014.5	9869.9

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
2.7	18940.4	13059.6	17675.8	9685.0
2.8	17989.2	12443.5	17396.0	9532.8
2.9	16930.9	11759.2	17005.1	9319.0
3.0	15752.8	10990.1	16642.3	9121.2
3.1	14731.3	10327.1	16359.6	8968.1
3.2	13905.6	9790.1	16103.1	8829.5
3.3	13273.9	9379.8	15869.6	8703.5
3.4	12792.7	9069.1	15635.8	8577.1
3.5	12397.6	8813.6	15410.8	8455.6
3.6	12073.4	8606.9	15212.6	8348.9
3.7	11800.0	8436.8	15044.8	8259.0
3.8	11538.1	8275.3	14888.2	8175.1
3.9	11269.6	8110.1	14735.6	8093.2
4.0	11006.0	7950.7	14590.6	8015.4
4.2	10530.4	7672.7	14305.9	7862.6
4.4	10082.0	7405.1	14254.2	7840.2
4.6	9741.4	7200.6	14877.5	8188.1
4.8	9551.8	7075.5	15040.0	8280.6
5.0	9456.1	6996.9	15165.4	8353.2
5.2	9355.9	6927.6	15009.7	8269.0
5.4	9233.8	6852.4	14844.4	8179.9
5.6	9328.0	6930.7	14725.9	8116.3
5.8	9741.9	7363.8	14641.4	8070.8
6.0	8637.1	7342.1	14443.3	7960.3
6.2	7508.8	6841.2	14190.6	7819.0
6.4	7404.0	6587.1	13944.6	7681.9
6.6	7563.0	6445.0	13731.3	7563.3
6.8	7777.2	6364.6	13539.1	7455.3
7.0	7948.8	6300.3	13333.8	7338.7
7.2	8039.9	6265.0	13127.8	7221.4
7.4	7991.6	6204.2	12919.3	7103.1

Time Seconds	Break Path No. 1(1) Mass lbm/sec	Break Path No. 1(1) Energy Thousand Btu/sec	Break Path No. 2(2) Mass lbm/sec	Break Path No. 2(2) Energy Thousand Btu/sec
7.6	7853.8	6118.4	12697.2	6978.1
7.8	7677.3	6023.6	12475.7	6854.6
8.0	7480.8	5923.4	12253.8	6731.4
8.2	7275.9	5818.5	12031.8	6608.5
8.4	7071.6	5709.1	11806.3	6483.7
8.6	6880.4	5600.4	11583.3	6360.6
8.8	6699.2	5489.7	11356.9	6235.6
9.0	6528.0	5377.1	11137.4	6114.7
9.2	6374.1	5275.8	10927.5	5999.3
9.4	6218.6	5165.7	10703.5	5876.4
9.6	6077.1	5057.6	10498.2	5764.1
9.8	5937.8	4947.5	10290.2	5650.4
10.0	5786.1	4826.6	10063.5	5526.1
10.2	5627.8	4694.1	9838.3	5402.9
10.4	5479.6	4555.0	9658.3	5296.1
10.6	5333.1	4404.6	9460.1	5155.4
10.8	5171.2	4238.1	9290.6	5005.7
11.0	4994.9	4056.1	9160.7	4858.2
11.2	4837.0	3883.3	9103.0	4740.5
11.4	4732.0	3743.6	9143.8	4681.0
11.6	4652.9	3629.3	9051.4	4568.6
11.8	4582.2	3535.7	8971.8	4473.9
12.0	4506.5	3455.5	8876.5	4379.7
12.2	4425.1	3384.4	8728.9	4267.2
12.4	4338.9	3320.1	8615.6	4176.9
12.6	4250.4	3262.5	8498.0	4089.7
12.8	4157.9	3209.4	8352.4	3989.6
13.0	4060.3	3160.0	8239.0	3903.7
13.2	3958.8	3115.7	8132.4	3821.0
13.4	3852.0	3075.5	7938.2	3698.4
13.6	3740.5	3042.2	7816.7	3610.0

Time Seconds	Break Path No. 1(1) Mass lbm/sec	Break Path No. 1(1) Energy Thousand Btu/sec	Break Path No. 2(2) Mass lbm/sec	Break Path No. 2(2) Energy Thousand Btu/sec
13.8	3623.6	3012.2	7726.1	3536.8
14.0	3502.4	2985.6	7572.3	3436.9
14.2	3373.9	2965.0	7330.9	3299.9
14.4	3239.1	2950.5	6904.7	3082.1
14.6	3100.5	2944.7	6504.7	2877.8
14.8	2941.7	2934.6	6327.1	2771.7
15.0	2691.8	2859.4	6000.4	2600.3
15.2	2412.2	2756.2	5418.4	2319.1
15.4	2201.2	2645.1	5092.6	2142.5
15.6	1956.1	2397.1	5145.2	2110.0
15.8	1755.7	2165.6	5560.3	2210.6
16.0	1584.9	1962.0	6003.9	2320.9
16.2	1434.0	1779.9	5875.2	2226.8
16.4	1300.5	1617.7	5266.8	1971.0
16.6	1186.0	1477.9	4659.5	1722.9
16.8	1064.8	1329.6	4213.7	1534.0
17.0	942.9	1179.1	3897.7	1391.6
17.2	843.4	1055.9	3588.8	1253.3
17.4	750.2	940.2	3263.7	1114.2
17.6	662.1	830.5	2960.7	988.5
17.8	581.5	730.0	2709.7	885.0
18.0	493.8	620.3	2438.3	779.6
18.2	409.1	514.3	2148.8	673.2
18.4	332.0	417.6	1833.9	563.8
18.6	259.5	326.8	1485.2	449.2
18.8	192.1	242.1	1101.0	328.6
19.0	128.0	161.6	682.1	201.8
19.2	80.0	101.2	260.9	77.0
19.4	30.7	39.0	0.0	0.0
19.6	0.0	0.0	0.0	0.0

1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 2: DEPS Break Reflood M&E Release – Minimum SI

Time Seconds	Break Path No. 1(1) Mass lbm/sec	Break Path No. 1(1) Energy Thousand Btu/sec	Break Path No. 2(2) Mass lbm/sec	Break Path No. 2(2) Energy Thousand Btu/sec
19.6	0.0	0.0	0.0	0.0
20.2	0.0	0.0	0.0	0.0
20.3	0.0	0.0	0.0	0.0
20.5	0.0	0.0	0.0	0.0
20.6	0.0	0.0	0.0	0.0
20.7	0.0	0.0	0.0	0.0
20.8	26.2	30.9	0.0	0.0
20.9	25.4	30.0	0.0	0.0
21.0	12.0	14.2	0.0	0.0
21.1	7.5	8.8	0.0	0.0
21.2	10.8	12.8	0.0	0.0
21.3	19.3	22.7	0.0	0.0
21.4	24.3	28.7	0.0	0.0
21.5	28.9	34.1	0.0	0.0
21.6	34.2	40.3	0.0	0.0
21.7	38.1	44.9	0.0	0.0
21.8	41.5	48.9	0.0	0.0
21.9	44.6	52.6	0.0	0.0
22.0	47.5	56.1	0.0	0.0
22.1	51.1	60.4	0.0	0.0
22.2	53.0	62.6	0.0	0.0
22.4	56.3	66.5	0.0	0.0
22.5	59.6	70.3	0.0	0.0
22.6	61.9	73.0	0.0	0.0
22.7	64.8	76.6	0.0	0.0
23.7	85.1	100.5	0.0	0.0
24.7	101.8	120.2	0.0	0.0
25.7	116.1	137.1	0.0	0.0
26.7	128.8	152.1	0.0	0.0
27.4	137.4	162.3	0.0	0.0

Time Seconds	Break Path No. 1(1) Mass lbm/sec	Break Path No. 1(1) Energy Thousand Btu/sec	Break Path No. 2(2) Mass lbm/sec	Break Path No. 2(2) Energy Thousand Btu/sec
27.7	140.1	165.5	0.0	0.0
28.7	150.7	178.1	0.0	0.0
29.7	220.6	260.9	1872.3	304.8
30.8	265.0	313.5	2532.2	423.3
31.8	264.9	313.4	2530.3	427.9
32.8	261.0	308.7	2481.1	422.9
33.3	258.9	306.3	2455.0	420.0
33.8	256.8	303.9	2428.9	417.2
34.8	252.9	299.1	2377.6	411.4
35.8	249.0	294.5	2327.4	405.7
36.8	245.3	290.1	2278.5	400.2
37.8	241.7	285.9	2230.8	394.7
38.8	238.3	281.8	2184.4	389.3
39.8	257.2	304.3	2484.5	406.2
39.9	256.9	303.9	2480.2	405.6
40.8	253.9	300.4	2442.1	401.0
41.8	250.7	296.6	2400.8	395.9
42.8	247.6	292.9	2360.5	390.9
43.8	244.7	289.4	2321.2	386.1
44.8	241.8	286.0	2282.8	381.3
45.8	239.0	282.6	2245.4	376.7
46.8	236.3	279.4	2208.8	372.1
47.1	235.5	278.5	2198.0	370.8
47.8	233.6	276.3	2173.1	367.7
48.8	231.1	273.3	2138.2	363.3
49.8	228.6	270.3	2104.0	359.0
50.8	226.2	267.5	2070.6	354.8
51.8	223.9	264.7	2037.8	350.7
52.8	221.6	262.0	2005.7	346.7
53.8	219.4	259.4	1974.3	342.7
54.8	217.2	256.8	1943.5	338.9

Time Seconds	Break Path No. 1[(1)] Mass lbm/sec	Break Path No. 1[(1)] Energy Thousand Btu/sec	Break Path No. 2[(2)] Mass lbm/sec	Break Path No. 2[(2)] Energy Thousand Btu/sec
55.0	214.8	254.0	1890.4	334.2
55.9	300.3	355.5	272.3	167.4
56.9	299.0	354.0	271.6	166.6
57.9	294.8	348.9	269.7	164.0
58.9	290.6	343.9	267.7	161.5
59.9	286.4	339.0	265.8	158.9
60.9	282.3	334.1	263.9	156.4
61.9	278.1	329.1	262.1	153.9
62.9	274.0	324.2	260.2	151.4
63.9	270.2	319.7	258.5	149.1
64.9	266.4	315.2	256.8	146.9
65.9	262.6	310.7	255.1	144.6
66.9	259.0	306.4	253.4	142.4
67.9	255.3	302.0	251.8	140.3
68.9	251.7	297.7	250.2	138.1
69.9	248.1	293.5	248.6	136.1
70.1	247.4	292.7	248.3	135.6
70.9	244.7	289.4	247.0	134.0
71.9	241.2	285.3	245.5	132.0
72.9	237.8	281.3	244.0	130.0
73.9	234.5	277.3	242.5	128.1
74.9	231.2	273.4	241.1	126.2
75.9	227.9	269.5	239.7	124.3
76.9	224.7	265.7	238.3	122.5
77.9	221.5	261.9	236.9	120.7
78.9	218.4	258.2	235.5	118.9
79.9	215.3	254.6	234.2	117.2
80.9	212.3	251.0	232.9	115.5
82.9	206.4	244.0	230.4	112.2
84.9	200.7	237.2	227.9	109.0
86.9	195.2	230.7	225.6	106.0

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
88.2	191.7	226.6	224.1	104.1
88.9	189.9	224.5	223.3	103.2
90.9	184.9	218.5	221.2	100.4
92.9	180.0	212.7	219.2	97.8
94.9	175.4	207.2	217.2	95.4
96.9	171.0	202.0	215.4	93.0
98.9	166.8	197.0	213.7	90.8
100.9	162.8	192.3	212.0	88.8
102.9	159.0	187.8	210.5	86.8
104.9	155.4	183.6	209.0	85.0
106.9	152.0	179.6	207.7	83.3
108.9	148.8	175.8	206.4	81.7
110.3	146.7	173.3	205.6	80.6
110.9	145.8	172.3	205.2	80.2
112.9	143.0	169.0	204.1	78.8
114.9	140.4	165.9	203.1	77.5
116.9	138.0	163.0	202.1	76.3
118.9	135.7	160.3	201.3	75.1
120.9	133.6	157.8	200.4	74.1
122.9	131.6	155.5	199.7	73.2
124.9	129.8	153.4	199.0	72.3
126.9	128.2	151.4	198.4	71.5
128.9	126.6	149.6	197.8	70.7
130.9	125.2	147.9	197.2	70.1
132.9	124.0	146.4	196.7	69.4
134.9	122.8	145.0	196.3	68.9
136.7	121.8	143.9	195.9	68.4
136.9	121.7	143.8	195.9	68.4
138.9	120.8	142.6	195.5	67.9
140.9	119.9	141.6	195.2	67.5
142.9	119.1	140.7	194.9	67.1

Time Seconds	Break Path No. 1[1] Mass lbm/sec	Break Path No. 1[1] Energy Thousand Btu/sec	Break Path No. 2[2] Mass lbm/sec	Break Path No. 2[2] Energy Thousand Btu/sec
144.9	118.5	139.9	194.6	66.8
146.9	117.9	139.2	194.4	66.5
148.9	117.4	138.6	194.2	66.2
150.9	116.9	138.1	194.0	66.0
152.9	116.5	137.6	193.9	65.8
154.9	116.2	137.2	193.7	65.7
156.9	115.9	136.9	193.6	65.5
158.9	115.7	136.6	193.5	65.4
160.9	115.5	136.4	193.4	65.3
162.9	115.3	136.2	193.3	65.2
164.9	115.2	136.0	193.3	65.1
165.9	115.1	135.9	193.3	65.1
166.9	115.1	135.9	193.2	65.0
168.9	115.0	135.8	193.2	65.0
170.9	115.0	135.8	193.2	65.0
172.9	115.0	135.8	193.2	64.9
174.9	115.0	135.8	193.1	64.9
176.9	115.0	135.8	193.1	64.9
178.9	115.0	135.9	193.1	64.9
180.9	115.1	135.9	193.2	64.9
182.9	115.2	136.0	193.2	64.9
184.9	115.3	136.2	193.2	65.0
186.9	115.4	136.3	193.2	65.0
188.9	115.5	136.4	193.2	65.0
190.9	115.7	136.6	193.3	65.1
192.9	115.8	136.8	193.3	65.1
194.9	116.0	137.0	193.4	65.2
196.8	116.1	137.2	193.4	65.2

1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Table 3: DEPS – Minimum Safety Injection Principal Parameters During Reflood

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total lbm/sec	Injection Accumulator lbm/sec	SI Spill lbm/sec	Enthalpy Btu/lbm
19.6	176.2	0.000	0.000	0.00	0.00	0.333	0.000	0.0	0.0	0.00
20.5	174.6	20.304	0.000	0.74	0.95	0.000	4358.000	4358.0	0.0	99.79
20.7	174.0	19.252	0.000	1.08	0.96	0.000	4328.600	4328.6	0.0	99.79
21.1	173.9	1.640	0.092	1.31	1.50	0.240	4256.900	4256.9	0.0	99.79
21.5	174.1	2.572	0.161	1.38	2.33	0.335	4194.700	4194.7	0.0	99.79
22.4	174.4	2.293	0.291	1.51	3.81	0.401	4082.900	4082.9	0.0	99.79
23.7	174.9	2.231	0.431	1.66	6.16	0.422	3918.400	3918.4	0.0	99.79
27.4	176.6	2.482	0.601	2.00	12.36	0.440	3528.400	3528.4	0.0	99.79
30.8	178.4	3.353	0.670	2.29	15.61	0.563	3103.100	3103.1	0.0	99.79
32.8	179.6	3.233	0.688	2.46	15.62	0.559	2955.400	2955.4	0.0	99.79
33.3	179.9	3.202	0.691	2.50	15.62	0.557	2922.600	2922.6	0.0	99.79
38.8	183.5	2.959	0.711	2.92	15.62	0.540	2601.400	2601.4	0.0	99.79
39.8	184.2	3.049	0.714	3.00	15.62	0.563	2926.000	2500.4	0.0	95.17
39.9	184.3	3.045	0.714	3.01	15.62	0.563	2921.000	2495.3	0.0	95.16
47.1	189.3	2.837	0.722	3.50	15.62	0.548	2598.800	2169.2	0.0	94.53
55.0	194.9	2.657	0.725	4.00	15.62	0.529	2256.900	1824.2	0.0	93.70
55.9	195.6	3.263	0.733	4.06	15.54	0.601	415.900	0.0	0.0	68.00
62.9	201.6	2.991	0.733	4.55	14.51	0.595	420.200	0.0	0.0	68.00
70.1	208.1	2.735	0.732	5.00	13.68	0.589	424.800	0.0	0.0	68.00
78.9	216.4	2.460	0.731	5.52	12.91	0.579	429.200	0.0	0.0	68.00

Time sec	Temp °F	Flooding Rate in/sec	Carry-over Fraction	Core Height ft	Down-Comer Height ft	Flow Fraction	Total lbm/sec	Injection Accumulator lbm/sec	SI Spill lbm/sec	Enthalpy Btu/lbm
88.2	225.0	2.210	0.729	6.00	12.37	0.567	432.700	0.0	0.0	68.00
98.9	233.8	1.978	0.727	6.51	12.03	0.553	435.300	0.0	0.0	68.00
110.3	241.6	1.793	0.725	7.00	11.92	0.537	437.200	0.0	0.0	68.00
124.9	249.8	1.637	0.725	7.57	12.05	0.521	438.600	0.0	0.0	68.00
136.7	255.5	1.561	0.726	8.00	12.30	0.513	439.300	0.0	0.0	68.00
152.9	262.3	1.506	0.729	8.57	12.75	0.506	439.700	0.0	0.0	68.00
165.9	267.1	1.485	0.733	9.00	13.17	0.505	439.800	0.0	0.0	68.00
182.9	272.6	1.475	0.738	9.56	13.73	0.505	439.800	0.0	0.0	68.00
194.9	276.1	1.473	0.742	9.94	14.14	0.506	439.800	0.0	0.0	68.00
196.8	276.6	1.473	0.743	10.00	14.20	0.506	439.800	0.0	0.0	68.00

Table 4: DEPS Break Post-Reflood M&E Release – Minimum SI

	Break Path No. 1(1)		Break Path No. 2(2)	
Time Seconds	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
196.8	156.2	194.3	286.0	97.1
201.8	152.0	189.0	290.2	95.6
206.8	155.3	193.2	286.9	96.9
211.8	154.6	192.3	287.6	96.8
216.8	154.5	192.2	287.7	96.6
221.8	153.8	191.3	288.4	96.6
226.8	153.7	191.2	288.5	96.3
231.8	153.0	190.3	289.2	96.3
236.8	152.9	190.2	289.3	96.1
241.8	152.1	189.2	290.1	96.1
246.8	152.0	189.1	290.2	95.8
251.8	151.2	188.1	291.0	95.8
256.8	151.0	187.9	291.2	95.6
261.8	150.2	186.9	292.0	95.6
266.8	150.0	186.7	292.2	95.4
271.8	149.2	185.6	293.0	95.4
276.8	149.0	185.4	293.2	95.2
281.8	148.2	184.3	294.0	95.1
286.8	147.9	184.0	294.3	95.0
291.8	147.6	183.7	294.6	94.8
296.8	146.8	182.6	295.4	94.8
301.8	146.5	182.2	295.7	94.6
306.8	146.1	181.8	296.1	94.4
311.8	145.2	180.6	297.0	94.4
316.8	144.9	180.2	297.4	94.3
321.8	144.5	179.7	297.7	94.1
326.8	144.0	179.2	298.2	94.0
331.8	143.1	178.0	299.2	94.0
336.8	146.4	182.1	295.8	95.1
341.8	145.8	181.4	296.4	95.0

Time Seconds	Break Path No. 1(1)		Break Path No. 2(2)	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
346.8	145.2	180.6	297.0	94.9
351.8	144.6	179.9	297.6	94.8
356.8	144.4	179.7	297.8	94.5
361.8	143.7	178.8	298.5	94.5
366.8	143.0	177.8	299.2	94.4
371.8	142.7	177.5	299.5	94.2
376.8	141.8	176.4	300.4	94.1
381.8	141.4	175.9	300.8	94.0
386.8	140.9	175.3	301.3	93.8
391.8	140.4	174.7	301.8	93.7
396.8	139.8	173.9	302.4	93.6
401.8	139.1	173.1	303.1	93.5
406.8	138.4	172.2	303.8	93.4
411.8	138.1	171.8	304.1	93.2
416.8	137.2	170.7	305.0	93.2
421.8	136.6	170.0	305.6	93.0
426.8	139.9	174.0	302.4	94.0
431.8	139.5	173.6	302.7	93.8
436.8	138.9	172.8	303.3	93.7
441.8	138.4	172.2	303.8	93.6
446.8	137.5	171.0	304.7	93.5
451.8	137.2	170.6	305.1	93.3
456.8	136.5	169.8	305.7	93.2
461.8	135.6	168.7	306.6	93.1
466.8	135.2	168.2	307.0	92.9
471.8	134.3	167.1	307.9	92.8
476.8	133.7	166.3	308.6	92.7
481.8	136.8	170.1	305.5	93.7
685.2	136.8	170.1	305.5	93.7
685.3	70.8	87.3	371.4	108.0
686.8	70.8	87.3	371.5	107.9

Time Seconds	Break Path No. 1[1]		Break Path No. 2[2]	
	Mass lbm/sec	Energy Thousand Btu/sec	Mass lbm/sec	Energy Thousand Btu/sec
1029.0	70.8	87.3	371.5	107.9
1029.1	64.0	73.6	378.2	31.0
1894.9	55.1	63.4	387.1	32.6
1895.0	55.1	63.4	71.5	11.2
3600.0	46.7	53.8	79.9	12.7
3600.1	34.5	39.7	92.1	6.3
4742.2	31.4	36.2	95.2	6.5

1. Path 1: M&E exiting from the steam generator side of the break.
2. Path 2: M&E exiting from the broken loop reactor coolant pump side of the break.

Summary of MSLB Results

a) Containment Integrity

The SLB containment integrity analyses were revised to address any applicable issues. The previously described input considerations that are applicable to SLB are:

- Containment Volume
- Initial Containment Temperature
- Initial Containment Pressure
- Initial Containment Relative Humidity
- Spray Droplet Size

The remaining changes were either specific to the LOCA analysis or remained conservative in the SLB analysis. The revised SLB containment results were evaluated against the previous EPU analyses used as input to the Reference 1 submittal. The limiting scenarios for peak containment pressure and temperature are shown in Figures 3 and 4. The reanalysis resulted in an increase of 1.1 psi with a peak of 53.4 psig. The revised results meet the design limit of 55 psig. The reanalysis resulted in an increase of 15.3°F to 279.2°F on the peak structural temperature which meets the design limit of 283°F. Therefore, the proposed EPU remains acceptable with the revised EPU SLB containment analysis.

b) Pressure/Temperature Curves

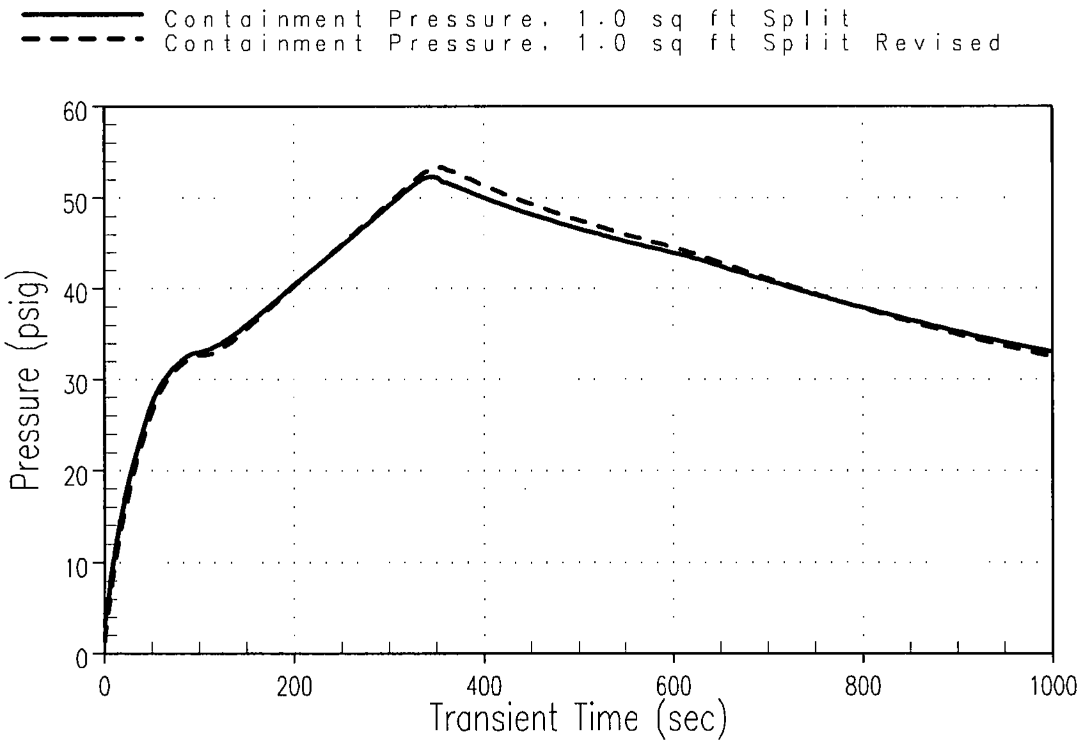

Figure 3: Turkey Point Units 3 and 4 Containment Pressure Comparison for the Limiting Pressure Case

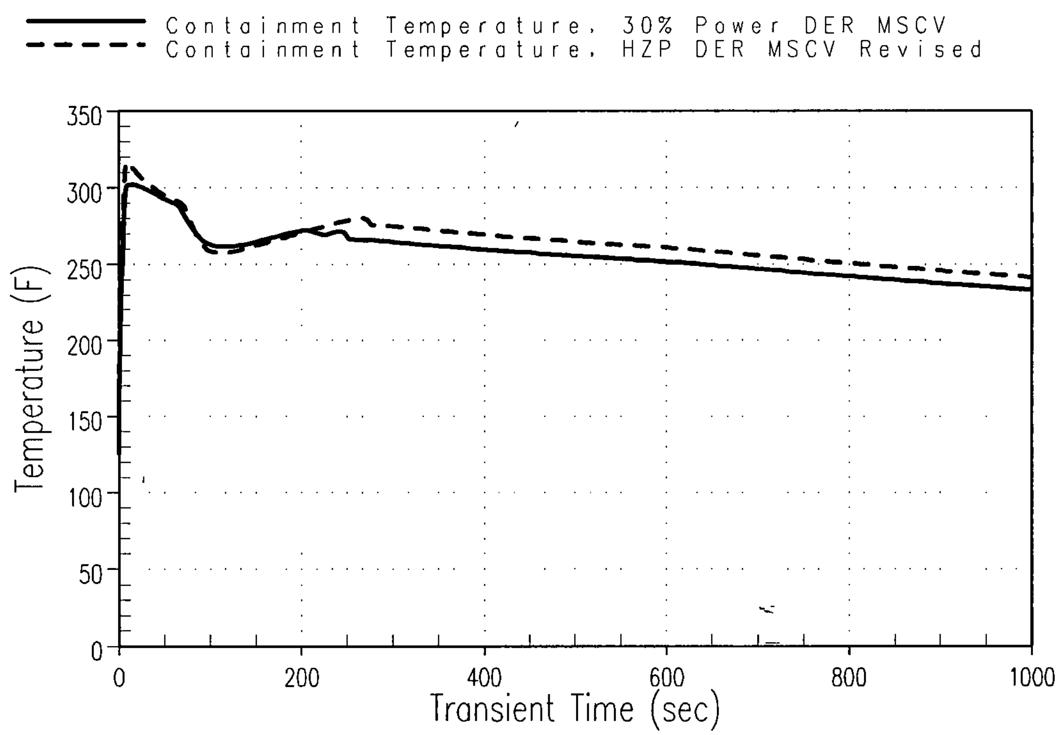

Figure 4: Turkey Point Units 3 and 4 Containment Temperature Comparison for the Limiting Temperature Cases

c) Mass and Energy Tables

Table 5: SLB Mass/Energy Releases from a Split Break Initiated from HZP with MSCV Single Failure

Time (sec)	Break Flowrate (lbm/sec)	Break Enthalpy (BTU/lbm)
0	0	0
0.005	742.17	895.17
0.01	1656.56	1062.08
0.06	2086.6	1193
0.155	2065.9	1192.57
0.47	2132.27	1193.32
0.81	2069.46	1193.57
7.22	1804.5	1197.72
16.03	1574.12	1200.8
54	1112.58	1204.58
58	953.84	1205.15
62	855.82	1205.01
66.5	767.18	1204.59
75.5	650.05	1203.56
84	585.74	1202.68
93	547.72	1202.04
110.5	514.24	1201.38
128.5	502.96	1201.15
274	475.18	1200.53
318	473.32	1200.53
333	454.57	1200.12
340	427.48	1199.54
342	408.99	1199.1
343	376.75	1200
344	352.88	1198.65
346	331.42	1196.7
349	315.53	1194.4
350	301.2	1193.77
353	235.42	1189.77
355	199.03	1187.49

Time (sec)	Break Flowrate (lbm/sec)	Break Enthalpy (BTU/lbm)
357	172.32	1184.02
359	151.71	1180.78
363	119.68	1176.92
366	100.02	1172.1
370	78.93	1167.81
372	70.7	1166.05
374	63.94	1163.85
378	54.08	1160.49
381	49.08	1159.19
385	44.59	1157.15
389	41.79	1156.13
393	40.13	1155.44
400	38.82	1154.91
599	38.57	1154.79
606	39.59	1154.81
610	35.04	1152.19
611	28.87	1151.89
612	18.64	1150.94
613	20.89	1150.34
616	18.11	1151.3
617	17.92	1150.45
618	18.36	1149.11
619	19.25	1145.96
620	17.62	1156.94
625	10.02	1159.57
627	4.65	1088.18
634	2.47	1173.42
640	2.33	1127.03
645	1.42	1163.02
650	0.82	1114.46
652	0.65	1118.6
653	0.0	0.0
1000	0.0	0.0

Sections of Attachment 4 of the LAR with Revised Analyses

2.5.4.3 Reactor Auxiliary Cooling Water Systems (Component Cooling Water System)
The containment analyses performed to correct the errors discovered in the EPITOME code increase the heat loads required to be dissipated by the CCW heat exchangers. This results in a reduction in the margin of heat exchanger fouling. The maximum allowable total tube resistance value (a measure of heat exchanger cleanliness) will continue to be administratively controlled but to a reduced value at EPU. Additionally, accident CCW temperatures increase slightly. For example, the CCW supply temperature increases for the LOCA analysis from the previously reported 157.8°F to 158.6°F. The credited CCW flows for accident conditions do not change from previous analyses. These changes do not alter the previous conclusions.

2.6.1 Primary Containment Functional Design
The input changes (including initial containment temperature and pressure, fan cooler actuation, and revised mass and energy releases) were included in the revised containment response. The results of the revised containment responses (delineated previously) show that Turkey Point Units 3 and 4 continue to meet all acceptance criteria.

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents
The input changes for the LOCA mass and energy releases (including plant specific decay heat model and use of an updated EPITOME code) resulted in revised LOCA mass and energy releases. These revised mass and energy releases in Tables 1, 2, 3, and 4 were provided as input into the containment response model.

2.6.3.2 Mass and Energy Release Analysis for Secondary System Pipe Ruptures
The input changes for the containment model impact the main steam line break mass and energy releases for split breaks, since these breaks model hi-1 and hi-2 protection signals for mitigation of these breaks. The revised mass and energy releases in Table 5 were provided as input into the containment response.

2.6.5 Containment Heat Removal
The containment heat removal margins for EPU were based on containment design conditions of 55 psig and 283°F and a maximum sump temperature of 244.8°F. Containment pressure and air temperature continue to be bounded by the containment design and maximum sump temperature decreased to 237.8°F. The control scheme for the Emergency Containment Coolers (ECCs) will be revised. If one of the two normally actuated ECCs fails to start, the third (swing) ECC will be automatically loaded on to the energized bus. This ensures that exactly two ECCs will operate automatically within one minute. The logic will be modified such that three ECCs cannot actuate during accident conditions and restores an original design feature. No TS change required.

Other Sections of Attachment 4 of the LAR Assessed for Impact

2.1.7 Protective Coating Systems (Paints) – Organic Materials

The containments at Turkey Point Units 3 & 4 have a variety of protective coatings on the steel and concrete structures as well as equipment and piping. These coatings were evaluated and the minimum qualifying test temperature and pressure was 307°F and 60 psig. These design conditions bound the revised LOCA containment peak temperature and pressure of 281.8°F and 53.9 psig. Due to EPITOME and containment free volume changes, the EPU radiation exposure is increasing from 2.256E+08 rads to 2.40E+08 rads, which is bounded by the limiting qualifying test exposure of 2.69E+08 rads. The pH of the containment sump is not impacted due the EPITOME and net free volume changes. Therefore, the coatings system parameters remain within the acceptance criteria for EPU.

2.2.4 Safety-Related Valves and Pumps

The Appendix J testing comprised of Type A, B, and C testing requirements have been evaluated based on the updated LOCA containment peak pressure of 53.9 psig at accident conditions, which is greater than the peak containment pressure post Main Steam Line Break of 53.4 psig. As the Appendix J program is being updated to reflect a design pressure of 55 psig at EPU, the change is bounded by previous recommendations. The PTN Technical Specification and the applicable PTN Appendix J Program procedures will be changed to address the value of Pa as being equal to 55 psig at EPU conditions. Further, the In-Service Testing and In-Service Inspection programs are not impacted. There is no impact due to EPITOME to the Motor Operated Valve / Air Operated Valve (MOV/AOV) Program. The results of the evaluations show that the design pressure of 55 psig continues to be bounded by the design pressure of the valves that use containment design pressure in their calculations. The increased LOCA containment peak temperature of 281.8°F has a very small effect on the starting voltage available to the MOVs. The starting voltages are sufficient at the new conditions.

2.3.1 Environmental Qualification of Electrical Equipment

The accident Temperature, Pressure, Radiation and Submergence values were changed to reflect the reduction in Containment Free Volume. No additional equipment was identified as requiring analysis due to this change in environmental conditions.

The LOCA containment peak pressure and temperature increased to 53.9 psig and 281.8°F. It was determined that there was sufficient design margin available to accommodate the changes. The resulting small change in temperature and pressure during the MSLB remain bounded by the LOCA design conditions and are therefore acceptable with no additional analysis required.

The submergence level increased only slightly (0.04 inch) and safety-related electrical equipment was reviewed against it for environmental qualification (EQ) impact. No new equipment was found that required additional analysis or relocation due to the change in accident flood level.

New radiation values for the accident were identified and accident conditions were reassessed to reflect changes in radiation levels due to the change in Containment Free Volume [Reference 5]. The EPU radiation exposure increased to 2.40E+08 rads of which 5.64E+07 rads is the accident EPU gamma dose and 1.46E+08 rads is the accident EPU beta dose with the remainder being the operating dose over the 60 year plant life. No new equipment was identified as a result of the change in radiation values resulting from the reduction in the Containment Free Volume.

Based on the review of the above changes it is concluded that all equipment in the EQ program remain qualified as is with no additional reviews or actions necessary.

2.6.6 Pressure Analysis for Emergency Core Cooling System Performance Capability

The containment input changes have no impact on the ASTRUM BE LBLOCA Analysis or on the pressure analysis for ECCS performance capability. The new containment input parameters being considered are all bounded by the values assumed in the EPU analyses presented in the LAR. Turkey Point Units 3 and 4 continue to maintain a margin of safety to the limits prescribed by 10 CFR 50.46 (including PCT).

2.7.7 Other Ventilation Systems (Containment)

HVAC Systems normal and emergency operating evaluations remain acceptable for EPITOME and containment net free volume changes with ECC control scheme modifications. If one of the two normally actuated ECCs fails to start, the third (swing) ECC will be automatically loaded on to the energized bus. This ensures that exactly two ECCs will operate automatically within one minute of receipt of an actuation signal. The logic will be modified such that three ECCs cannot actuate during accident conditions and restores an original design feature. No TS change required.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

The design input changes discussed above were evaluated for impact on the AST radiological dose consequence analyses. The net impact of all these changes on the onsite radiological dose consequences for the control room, the technical support center (TSC), the low population zone (LPZ), and the exclusion area boundary (EAB) was determined to be relatively small and the AST radiological doses as approved for LAR-196 remain bounding.

Conclusion

In conclusion, both the LOCA and MSLB containment integrity analyses were revised to address the applicable issues associated with the accident mass and energy release analyses identified by Westinghouse as well as identified changes to the containment net free volume. Changes to some design inputs including reduction of the maximum TS allowable containment internal pressure during MODES 1-4 and modification of the ECC accident initiation sequence were necessary to accommodate the increases in the mass and energy releases and the smaller containment net free volume. The effect of the revised containment analyses were evaluated against the previous EPU evaluations submitted in Reference 1. FPL has determined that the previous EPU evaluations remain valid. Therefore, the proposed EPU remains acceptable with the revised EPU LOCA and MSLB containment integrity analyses.

Technical Specification (TS) Changes

TS 3.6.1.4 Containment Systems – Internal Pressure

Current TS

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +3 psig.

Proposed TS

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and **+1** psig.

Basis for Change: Additional margin between the design basis containment pressure and the peak design basis containment pressure was required as a result of the EPITOME M&E/Containment Analysis error. A lower maximum primary containment internal pressure of +1 psig is used as an initial condition in both the LOCA and MSLB containment integrity analyses and results in lower post accident peak containment pressures. The lower containment pressure is more restrictive and will assure that the accident peak containment pressure remains within design limits.

See Figure 5a for TS change.

TS 3.4.6.1 Reactor Coolant System Leakage – Leakage Detection Systems

Current TS

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System...

Action:

a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

...

(4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.

Proposed TS

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System...

Action:

a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

...

(4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.**

****Instrument Air Bleed Valves may be opened intermittently under administrative controls.**

Basis for Change: Isolation of the containment purge supply, purge exhaust, and instrument air bleed flow paths are required due to the inoperability of both containment particulate and gaseous radiation monitors but intermittent opening of the instrument air bleed valves under administrative control is desirable during the 7 day allowed outage time in order to maintain the containment internal pressure within the limits of TS 3.6.1.4. This action is consistent with the provision in the Notes of NUREG-1431, Standard Technical Specifications (STS) for Westinghouse Plants, Section 3.6.3 on Containment Isolation Valves which state that penetration flow paths, except for the purge supply and exhaust flow paths, may be unisolated intermittently under administration controls. The associated TS Bases explain that these administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. In this case, the valve controls are located in the control room such that the control room operator can directly monitor the activity and take action directly to close the valve as required. Moreover, these valves automatically isolate upon receipt of either a containment high radiation signal or safety injection signal, the latter of which is operable during this period. As indicated in the STS TS Bases Section B3.3.6 on Containment Purge and Exhaust Isolation Instrumentation, the containment radiation monitors act as backup to the safety injection signal to ensure closing of the valves. They are the primary means of automatically isolating containment in the event of a fuel handling accident and, for the LOCA and Rod Control Cluster Assembly (RCCA) Ejection Design Basis Accidents, they are credited for initiating containment ventilation (and control room) isolation. However, the differences between the signal initiation times for these latter two events result in little or no impact on dose consequences.

See Figure 5b for TS change. See Figure 6a for TS Bases change:

TS Table 3.3-4 Radiation Monitoring Instrumentation for Plant Operations, Action 26
- **Function 1a, Containment Atmosphere Radioactivity-High, and**
- **Function 1b, RCS Leakage Detection Particulate or Gaseous Radioactivity-High**

Current TS

Action 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

...4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.

Proposed TS

Action 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

...(4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.**

**Instrument Air Bleed Valves may be opened intermittently under administrative controls.

Basis for Change: See basis discussion under TS 3.4.6.1 above.

See Figure 5c for TS change. See Figure 6b for TS Bases change:

Environmental Evaluation

The proposed changes do not affect the conclusions of the original environmental evaluation provided with the EPU LAR No. 205 submittal via letter L-2010-113 dated October 21, 2010.

No Significant Hazards Determination

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazard if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

FPL proposes to revise Turkey Point Units 3 and 4 Technical Specification Limiting Condition for Operation (LCO) Requirement 3.6.1.4 for Containment Internal Pressure to reduce the maximum allowable containment internal pressure from 3 psig to 1 psig in order to provide for additional margin between the design basis containment pressure and the peak design basis containment pressure indicated in the LOCA design basis accident analysis. The additional margin was desired to accommodate the revised LOCA containment analysis results due to an EPITOME Computer Code M&E/Containment Analysis error. In addition, changes to the Technical Specification 3.4.6.1 Action (a).4 and Table 3.3-4 Action 26.(4) are proposed to allow the instrument air bleed valves to be intermittently opened under strict administrative controls in order to maintain containment internal pressure within the specified limits of TS 3.6.1.4.

FPL has reviewed this proposed license amendment for FPL's Turkey Point Units 3 and 4 and determined that its adoption would not involve a significant hazards consideration. The bases for this determination are:

The proposed amendment does not involve a significant hazards consideration for the following reasons:

1. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   The proposed change to TS 3.6.1.4 will narrow the allowable range of values for the containment internal pressure during Modes 1-4 by reducing the maximum allowable containment internal pressure from 3 psig to 1 psig. This will result in a reduction in both the initial condition assumed in the design basis accident analyses and in the calculated post-LOCA peak containment pressure to help assure that adequate margin exists between the peak accident containment pressure and the design basis containment pressure of 55 psig. The maximum containment internal pressure during normal operation is not an initiator of any analyzed accident and its reduction will only act to decrease the containment analysis results, i.e., containment pressure, and associated dose consequences.

   The proposed changes to TS 3.4.6.1 and TS Table 3.3-4 will allow the control room operator to intermittently reopen the instrument air purge isolation valves under strict administrative controls in order to maintain the containment pressure within the limits of TS 3.6.1.4 in the event that the direct flow path to the environment has been isolated due to inoperability of both the containment particulate and gaseous radioactivity monitors. The instrument air bleed valves are normally maintained in the open position and do not constitute an initiator of any analyzed accident. Although containment high radiation isolation inputs to the containment ventilation isolation signal would be inoperable during these action statements, the other automatic input signals on safety injection would still be operable in the event of a LOCA or RCCA ejection accident. The differences in the two signal receipt times for these events are inconsequential relative to the calculated dose consequences. In addition, although not credited in the RCCA ejection analysis, the control room operator responsible for monitoring containment pressure while the valves were unisolated would close the valves upon initial indication of the event.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   As indicated above, the maximum containment internal pressure during normal operation is not an accident initiator and its reduction is not an initiator. Similarly, the proposed change to TS 3.4.6.1 and Table 3.3-4 to allow intermittent re-opening of the instrument air bleed valves under strict administrative control during the TS 3.4.6.1 allowed outage time does not act as an initiator or contribute to the initiation of any accident. The proposed change does not significantly affect mitigation strategies as the containment ventilation isolation signal is backed up by both the operable automatic safety injection signal and manual initiation by the control room operator responsible for monitoring conditions in containment while the valves are unisolated.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed amendment does not involve a significant reduction in the margin of safety.**

   The proposed change to TS 3.6.1.4 does not result in any challenge to fuel, reactor pressure vessel, or containment integrity and actually are intended to increase the overall margin for the containment. Similarly, the proposed changes to TS 3.4.6.1 and Table 3.3-4 do not result in any challenge to fuel, reactor pressure vessel, or containment integrity and do not result in any discernible increase in the analyzed radiological dose consequences.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above discussion, FPL has determined that the proposed change does not involve a significant hazards consideration.

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 – Containment and Ventilation (SCVB) Request for Additional Information - Round 1," Accession No. ML110950084, April 1, 2011.
3. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-084), "Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Containment and Ventilation Issues," Accession No. ML11119A135, April 28, 2011.
4. WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983.
5. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-223), "Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Electrical Engineering Branch Issues," July 7, 2011.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and ~~+3~~ +1 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

Figure 5a – TS 3.6.1.4 Containment Internal Pressure

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment Sump Level Monitoring System is OPERABLE;

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;

3) A Reactor Coolant System water inventory balance is performed at least once per 8* hours except when operating in shutdown cooling mode; and

4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.**

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

* Not required to be performed until 12 hours after establishment of steady state operation.

**Instrument Air Bleed Valves may be opened intermittently under administrative controls.

Figure 5b – TS 3.4.6.1 RCS Leakage Detection Systems

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

\# Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(F)}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

1) A Containment sump level monitoring system is OPERABLE,

2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,

3) A Reactor Coolant System water inventory balance is performed at least once per 8*** hours except when operating in shutdown cooling mode, and

4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.****

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

*** Not required to be performed until 12 hours after establishment of steady state operation.

****Instrument Air Bleed Valves may be opened intermittently under administrative controls.

Figure 5c – Radiation Monitor Action Statement

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TECHNICAL SPECIFICATION BASES

3/4.4.5 (Cont'd)

References

1. NEI 97-06, Steam Generator Program Guidelines
2. 10 CFR 50 Appendix A, GDC 19
3. 10 CRF 100
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Draft Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes, August 1976
6. EPRI Pressurized Water Reactor Steam Generator Examination Guidelines
7. 10 CFR 50.67, Accident source term

3/4.4.6 Reactor Coolant System Leakage

3/4.4.6.1 Leakage Detection Systems

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary to the containment. The containment sump level system is the normal sump level instrumentation. The Post Accident Containment Water Level Monitor - Narrow range instrumentation also functions as a sump level monitoring system. In addition, gross leakage will be detected by changes in makeup water requirements, visual inspection, and audible detection. Leakage to other systems will be detected by activity changes (e.g., within the component cooling system) or water inventory changes (e.g., tank levels).

INSERT 1 →

Background

Components that contain or transport the coolant to or from the reactor core make up the Reactor Coolant System (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant Leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational Leakage LCO is to limit system operation in the presence of Leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

W2003:DPS/mr/cls/cls

Figure 6a – TS 3.4.6.1

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TECHNICAL SPECIFICATION BASES

3/4.3.3 Monitoring Instrumentation

3/4.3.3.1 Radiation Monitoring for Plant Operations

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that conditions indicative of potential uncontrolled radioactive releases are monitored and that appropriate actions will be automatically or manually initiated when the radiation level monitored by each channel reaches its alarm or trip setpoint.

INSERT 1 →

3/4.3.3.2 Movable Incore Detectors

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976 or in the Westinghouse Single Point Calibration Technique, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 Accident Monitoring Instrumentation

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident, May 1983 and NUREG-0737, Clarification of TMI Action Plan Requirements, November 1980.

Action c states that separate Action entry is allowed for each Instrument. This Action has been added for clarification. The Actions of this Specification may be entered independently for each Instrument listed on Table 3.3-5. Allowable outage times of the inoperable channels of an Instrument will be tracked separately for each Instrument starting from the time the Action was entered for that Instrument.

W2003:DPS/mr/ds/ds

Figure 6b – TS Table 3.3-4 Action 26

INSERT 1

"In Modes 1-4, with both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, the isolation valves in the Containment Purge Supply and Exhaust and Instrument Air Bleed flow paths are required to be maintained closed in order to allow continued operation for up to 7 days. A note permits the instrument air flow path to be opened under administrative control in order to maintain the containment internal pressure within specified limits since it is relatively small in size and easily isolated either automatically by a safety injection signal or manually by the operator monitoring containment conditions while the valves are open."