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ppl TM

January 18, 2002

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington, DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
SUPPLEMENT NO. 4 TO PROPOSED AMENDMENT
NO. 241 TO LICENSE NPF-14 AND PROPOSED
AMENDMENT NO. 206 TO LICENSE NPF-22:
REQUEST FOR A ONE TIME DEFERRAL OF THE
TYPE A CONTAINMENT INTEGRATED LEAK
RATE TEST (ILRT) AND THE DRYWELL-TO-SUPPRESSION
CHAMBER BYPASS LEAKAGE TEST SR 3.6.1.1.2
PLA-5424**

**Docket No. 50-387
and 50-388**

Reference:

1. *PLA-5342, G. T. Jones (PPL) to USNRC, "Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT)", dated July 30, 2001.*
2. *PLA-5361, R. G. Byram (PPL) to USNRC, "Supplement to Proposed Amendment No 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT)", dated September 7, 2001.*
3. *Letter, R. G. Schaaf (USNRC) to R. G. Byram (PPL), "Susquehanna Steam Electric Station, Units 1 and 2 - Request for Additional Information Re: Deferral of Containment Integrated Leak Rate Testing (TAC Nos. MB2894 and MB2895)", dated October 5, 2001.*
4. *PLA-5380, R. G. Byram (PPL) to USNRC, "Supplement No. 2 to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT)", dated October 16, 2001.*
5. *PLA-5408, R. G. Byram (PPL) to USNRC, "Supplement No. 3 to Proposed Amendment No. 241 to License NPF-14 and Proposed Amendment No. 206 to License NPF-22: Request for a One Time Deferral of the Type A Containment Integrated Leak Rate Test (ILRT) and the Drywell-to-Suppression Chamber Bypass Leakage Test SR 3.6.1.1.2", dated December 5, 2001.*

A017

The purpose of this letter is to support the NRC's continuing review of our requests for one time deferral of the Type A Containment Integrated Leak Rate Test (ILRT) and the Drywell-To-Suppression Chamber Bypass Leakage Test SR 3.6.1.1.2.

On July 30, 2001, PPL Susquehanna, LLC (PPL) proposed revisions to the Susquehanna Steam Electric Station Units 1 and 2 Technical Specifications for NRC review. The revisions, if approved, would allow a one time deferral of the Type A Containment Integrated Leakage Rate Test (ILRT).

The PPL submittal (Reference 1) included a commitment to provide a risk assessment of the proposed action, which was forward to the NRC on September 7, 2001 (Reference 2). The NRC subsequently issued a request for additional information on October 5, 2001 (Reference 3), to which PPL responded in a letter dated October 16, 2001 (Reference 4).

The need for further information was identified during teleconferences between NRC and PPL on November 14, 25, and 26, 2001. PPL provided the additional information in a letter dated December 5, 2001, (Reference 5). Subsequently, PPL was requested to revise input assumptions in the PPL calculation EC-RISK-1081, Revision 1 contained in Reference 4. This letter provides the revised Appendix E, "Containment Fragility Calculation" in Attachment 1 which addresses postulated corrosion from the uninspectable side of the liner plate.

The revised Appendix E risk assessment concludes:

- The total integrated plant risk for those accident sequences influenced by concealed corrosion, given a change from 10-15 year test interval increases by 0.47%. This value is an insignificant increase in risk.

- Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as increases in core damage frequency (CDF) below 1E-6/year and increases in Large Early Release Frequency (LERF) below 1E-7/year. Since the concealed corrosion does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from concealed corrosion from 10 year test interval to a 15 year test interval is 1.30E-09/year. Because guidance in Regulatory Guide 1.174 defines very small changes in LERF as below 1E-07/year, increasing the ILRT interval from 10 to 15 years is not considered risk significant.

Finally, Attachment 2 to this letter provides an updated No Significant Hazards Consideration (NSHC) Evaluation that reflects consideration of the supplemental information provided in this response. There is no effect on the previous determination that this revision does not:

- Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
- Create the possibility of a new or different kind of accident from any accident previously analyzed; or
- Involve a significant reduction in a margin of safety.

If you have any questions on this submittal, please contact Mr. M. H. Crowthers at (610) 774-7766.

Sincerely,

B. L. Shriver

Attachments (2)

copy: NRC Region I
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. D. S. Collins, NRC Project Manager

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

SUPPLEMENT NO. 4 TO PROPOSED AMENDMENT NO. 241 TO LICENSE NPF-14: ONE TIME DEFERRAL OF THE CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT) AND THE DRYWELL-TO-SUPPRESSION CHAMBER BYPASS LEAKAGE TEST SR 3.6.1.1.2 UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files supplement No. 4 to Proposed Amendment No. 241 in support of a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment involves a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
By:

Bryce L. Shriver

B. L. Shriver
Vice-President – Nuclear Site Operations

Sworn to and subscribed before me this 18 day of January, 2002.

Sandra K. Lines

Notary Public

Notarial Seal
Sandra K. Lines, Notary Public
Salem Twp., Luzerne County
My Commission Expires Oct. 24, 2005
Member, Pennsylvania Association Of Notaries

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

SUPPLEMENT NO. 4 TO PROPOSED AMENDMENT NO. 206 TO LICENSE NPF-22: ONE TIME DEFERRAL OF THE CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT) AND THE DRYWELL-TO-SUPPRESSION CHAMBER BYPASS LEAKAGE TEST SR 3.6.1.1.2 UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files supplement No. 4 to Proposed Amendment No. 206 in support of a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment involves a revision to the Susquehanna SES Unit 2 Technical Specifications.

PPL Susquehanna, LLC
By:

Bryce L. Shriver

B. L. Shriver
Vice-President – Nuclear Site Operations

Sworn to and subscribed before me
this 18 day of January, 2002.

Sandra K Lines

Notary Public

Attachment 2 to PLA-5424

Revised No Significant Hazards Considerations Evaluation

(Units 1 & 2)

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION (REVISED)

PPL Susquehanna, LLC has evaluated the proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92 (c) a proposed amendment to an operating license involves no significant hazards considerations if operation of the facility with the proposed amendment would not:

- Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
- Create the possibility of a new or different kind of accident from any previously analyzed; or
- Involve a significant reduction in a margin of safety.

PPL Susquehanna, LLC proposes to:

Revise SSES Unit 1 Technical Specifications (TS) 5.5.12, Containment Leakage Rate Testing Program," by revising the end of the first paragraph and adding Section a. as follows:

... September 1995, as modified by the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 4, 1992 Type A test shall be performed no later than May 3, 2007.

Revise SSES Unit 2 Technical Specifications (TS) 5.5.12, Containment Leakage Rate Testing Program," by revising the end of the first paragraph and adding Section a. as follows:

... September 1995, as modified by the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the October 31, 1992 Type A test shall be performed no later than October 30, 2007.

NUCLEAR ENGINEERING CALCULATION / STUDY COVER SHEET and DCS TRANSMITTAL SHEET	1. Page 1 of 95 Total Pages 96

>2. TYPE: CALC >3. NUMBER: EC-RISK-1081 >4. REVISION: 2

5. TRANSMITTAL#: ______ *>6. UNIT: 3 *>7. QUALITY CLASS: N

>9. DESCRIPTION: Risk Impact Assessment of Extending Containment ILRT Interval *>8. DISCIPLINE: 3

SUPERSEDED BY: ______

10. Alternate Number: None 11. Cycle: NA

12: Computer Code or Model used: NA Fiche ☐ Dis ☐ Am't ______

13. Application: None

*>14 *Affected Systems:* 059 , , , , ,

* If N/A then line 15 is mandatory

**>15. NON-SYSTEM DESIGNATOR: RISK RADN

**If N/A then line 14 is mandatory

16. Affected Documents: None

☐ Lic. Doc Change Req'd

17. References: See body of calculation

18. Equipment / Component #: None

19. DBD Number: None

	Print Name	Signature
>20. PREPARED BY	Michael A Adelizzi	Michael A Adelizzi 1/15/02
>21. REVIEWED BY	Jack G Refling	Jack G. Refling 01/15/2002
>21A. VERIFIED BY	NA	
>22. APPROVED BY	F. G. Butler	F G Butler 01/15/02
>23. ACCEPTED BY PP&L / DATE		Signature / DATE

TO BE COMPLETED BY DCS

NR-DCS SIGNATURE/DATE

ADD A NEW COVER PAGE FOR EACH REVISION
FORM NEPM-QA-0221-1, Revision 5, Page 1 of 2, ELECTRONIC FORM

* Verified Fields
> REQUIRED FIELDS

ENGINEERING CALCULATION STUDY
REVISION DESCRIPTION SHEET

REVISION NO: 2	CALCULATION NUMBER: EC-RISK-1081

This form shall be used to record the purpose or reason for the revision, indicate the revised pages and / or affected sections and give a short description of the revision. Check (x) the appropriate function to add, replace or remove the affected pages.

Revised Pages	Affected Sections	Add	Rpl	Rmv	Description / Purpose of Revision
79-97	Appendix E		X		Changed failure pressure from 124 psia to 62 psia in Containment
					Fragiltiy Calculation (Appendix E). Replace Appendix E entirely.
					No revision bars are used on new pages starting with page 79

REVISION TYPE: (check one)

☐ SUPERSEDED BY ____________

☐ FULL REVISION

☒ PAGE FOR PAGE

Appendix E

Containment Fragility Calculation

Summary

The risk increase in extending ILRT from 10 years to 15 years on the total integrated plant risk due to corrosion of the containment liner from the concealed surface is 0.47%. The increase in LERF is 1.30E-09 / year. Both of these increases are not significant.

A. Objective

Provide a risk impact assessment on containment degradation due to corrosion of the containment liner from the concealed surface (concealed corrosion). This risk assessment is required as part of justification for deferring the plant's Type A Test interval from 10 to 15 years. This risk assessment is performed separate from the Type A Test assessment in the main body of the calculation. The Type A test assessment contained in the main body of the calculation is comparable to the analysis in NUREG – 1493, The analysis contained in this Appendix is not comparable to the NUREG since this analysis assesses the risk of liner plate corrosion from the concealed surface that the NUREG did not. The risk assessment will be performed in accordance with the guidelines set forth in NEI 94-01 (1), the methodology used in EPRI TR-104285 (2), and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a licensee request for changes to a plant's licensing basis, Reg. Guide 1.174 (3).

Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. This calculation will demonstrate that the increased risk to the public (person-rem / year) is insignificant. This calculation will demonstrate per Reg. Guide 1.174 that the change in risk increases CDF less than 1E-06/year and increases LERF less than 1E-07/year.

The results and findings from the SSES Individual Plant Examination (IPE) (4) are used for this risk assessment calculation.

B. Conclusion

The conclusions regarding the assessment of the plant risk associated with concealed corrosion are as follows:

1. The risk assessment associated with concealed corrosion from 10 years to 15 years predicts a slight increase in risk when compared to that estimated from current requirements. The change in risk for Class 7 as measured by person-rem/year increases by 0.55%. Also, the total integrated plant risk for those accident sequences influenced by concealed corrosion, given the change from 10 year test interval to a 15 year test interval increases by 0.47%. This value is an insignificant increase in risk.

2. Reg. Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1.0E-06/year and increases in LERF below 1.0E-07/year. Since the concealed corrosion does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from concealed corrosion from 10 year test interval to 15 year test interval is 1.30E-09 / year. Since guidance in Reg. Guide 1.174 defines very small changes in LERF as below 1.0E-07/yr, increasing the ILRT interval to 15 years is therefore not risk significant.

	Risk Results		
	Base line - 3 year interval	10 year interval	15 year interval
Total (person-rem per year)	1.6297	1.6409	1.6486
Increase over 3 year		0.69%	1.16%
Increase over 10 year			0.47%
Class 7 (person-rem per year)	1.4810	1.4928	1.5010
Class 7 Increase over 3 year		0.8%	1.3%
Class 7 Increase over 10 year			0.55%

	LERF Results (per year)		
	Base line - 3 year interval	10 year interval	15 year interval
LERF (class 3b)	9.43E-09	9.43E-09	9.43E-09
LERF (class 7)	2.36E-07	2.38E-07	2.39E-07
Total LERF	2.46E-07	2.48E-07	2.49E-07
Increase over 3 year		1.88E-09	3.18E-09
Increase over 10 year			1.30E-09

C. Assumptions

1. Same as Section 3.0.
2. Type A ILRT and Drywell to Suppression Chamber Bypass Test are performed at 3 year intervals. This assumption is used in order to assess the risk impact based on concealed corrosion alone.
3. The Type A ILRT will fail due to 100% corrosion. The ILRT pressure is 59.7 to 62.7 psia. This analysis uses 62 psia in the pressure range because the data for the analysis existed for this pressure. The result is not sensitive if pressure changes by a few psi.

D. Method

The following steps are used to perform the analysis:

1. Concealed corrosion discussion
2. Risk assessment of concealed corrosion

Concealed Corrosion Discussion

Corrosion damage has been found in approximately one-third of existing nuclear power plant containments. (25) Most of the corrosion found to date has started on the visible surface of the containment. Recent reports at Brunswick 2 (April 27,1999) and North Anna 2 (September 23,1999) identified 100% corrosion through the containment liner (27,28). The corrosion initiated from the concealed side of the liner due to debris left during construction. Both of these containments are similar to Susquehanna because they are steel lined concrete containments (26). Like Susquehanna, the steel liner is flush to the concrete structure.

The SSES primary containment is inspected in accordance with the requirements of ASME Section XI Subsection IWE and IWL. These visual inspections include the interior liner and the exterior concrete surfaces. As of April 2001, all inspections of both Unit 1 and Unit 2 primary containment for the first inspection period are complete and no degradation was identified. These inspections provide reasonable assurance that corrosion on the visible surface will be identified and corrected before containment strength is affected. However, corrosion that starts on the concealed surface may not be found before the damage affects containment strength.

In 1987 (30), the NRC sponsored a test of a 1:6 scale reinforced concrete containment model. A steel liner was incorporated into the model to provide a leak tight pressure boundary. For the overpressurization test, no significant leakage was detected until the pressure reached 135 psig. At 135 psig, the leakage was measured at 11% mass per day. The test was terminated at 145 psig when leakage exceeded 5000% mass per day. The scale model liner had no corrosion. This scale model is similar to the Susquehanna containment design of a reinforced concrete containment with a steel liner attached to the concrete. This test validates the Susquehanna containment ultimate strength of 140 psig and indicates that a non-degraded containment will remain leak tight almost until failure.

Chapter 7 of NUREG/CR-6706 (25) analyzes typical reinforced concrete containment using finite element techniques. . This analysis includes degradation due to corrosion. The analysis uses 0%, 10%, 25%, and 50% degradation. With no degradation, the exterior hoop reinforcing bars yielded at 75 psia, the interior hoop reinforcing bars yielded at 110 psia, and the liner yielded at 150 psia. This analysis shows that the exterior concrete cracks first, followed by the interior concrete, and finally the steel liner.

NUREG/CR-6706 (25) performs analyses with 3 different values of degradation in three locations. The conclusion is that degradation of the liner attached to the concrete with studs can degrade the ultimate strength of the containment by 20% when the liner is corroded by 50%.

The SSES containment liner is considered the leak proof membrane. The leakage rate out of containment with 100% degradation in the liner is not known from analysis or scale testing. However, when Brunswick 2 found 100% degradation of the containment liner, they conducted an as-found local leak rate test on the liner defects. The test was conducted at 55 psia. The total leakage through the 3 defects was 168 SCFH. This leakage was added to their containment leakage summation. The total leakage remained below the maximum allowed leakage rate (La) of 266.3 SCFH. Brunswick 2 concluded that primary containment integrity was maintained with the 100% degradation. (31)

However, to be conservative, we will analyze the effects of containment failure at 62 psia if 100% degradation exists. The leakage rate of a failed containment will be assumed to be 100 La.

Risk Assessment of Concealed Corrosion

The Severe Accident (Class 7) frequency in the original calculation was based on containment failure at 155 psia (140 psig). With 100% degradation of the liner present, containment will fail at a lower pressure. Based on previous discussion, containment will assume to fail at 62 psia due to 100% degradation of the liner to provide an upper bound on the risk impact of the proposed change.

The following steps are used for the risk assessment:

1. Determine sequences that are affected by lower containment strength and recalculate PRA.
2. Calculate risk for 3 year concealed corrosion test interval.
3. Calculate risk for 10 year concealed corrosion test interval.
4. Calculate risk for 15 year concealed corrosion test interval.
5. Calculate change in LERF

Step 1 - Determine sequences that are affected by lower containment strength and recalculate PRA

The IPE (4) determined that containment fails due to overpressure during station blackout sequences and loss of decay heat removal sequences.

The PRA has inputs for recovery of offsite power, recovery of diesel generator, and repair of pump for decay heat removal.

Time of containment failure is important for this analysis. The increased risk is due to containment failure occurring earlier when corrosion is present. Calculate the revised containment failure time as follows.

Time to reach 155 psia

The containment with no corrosion fails at 155 psia. The elevated pressure is due to decay heat from the core and is the result of water vapor. The saturated temperature for 155 psia is 361 F. The bulk temperature of the suppression pool is 353 F because 8 F difference accounts for temperature stratification in the suppression pool.

1. Calculate the mass of steam added to the suppression pool as follows (Ref 32, p.157):

Mass of steam added (lbm) =

$$\frac{\text{Mass of suppression pool (initial)} \times \text{(final enthalpy of water – initial enthalpy of water)}}{\text{(final enthalpy of steam – final enthalpy of water)}}$$

Mass of suppression pool (initial) = 7,940,000 lbm
(This represents 7,600,000 lbm initially plus 340,000 lbm from the initial blowdown of the RPV to the suppression pool. This blowdown is included to account for a rapid depressurization associated with the HCTL.)

Final enthalpy of water = 325 BTU/lbm
Initial enthalpy of water = 106 BTU/lbm
(The suppression pool water temperature after the initial blowdown is 138 F)
Final enthalpy of steam = 1194 BTU/lbm

Mass of steam added (lbm) =

$$\frac{7{,}940{,}000 \text{ lbm} \times (325 - 106) \text{ BTU/lbm}}{(1194\text{-}325) \text{ BTU/lbm}} = 2.00\text{E+06 lbm}$$

2. Calculate the energy of steam added to the suppression pool as follows:

Total energy added to pool (MW-sec) =

Mass of steam added x (Final enthalpy of steam – ((Final enthalpy of water + Initial enthalpy of water) x 0.5)) x 0.00106 MW-sec/BTU

Mass of steam added = 2.00E+06 lbm
Final enthalpy of steam = 1194 BTU/lbm
Final enthalpy of water = 325 BTU/lbm
Initial enthalpy of water = 106 BTU/lbm

Total energy added to pool (MW-sec) =

= 2.00E+06 lbm x (1194 – ((325 + 106) x 0.5)) BTU/lbm x 0.00106 MW-sec/BTU
= 2.08E+06 MW-sec

Time to 155 psia is calculated based on Eric Haskin Tabular Decay Heat (Ref 32, p.351)

Cumulative power of 2.08E+06 MW-sec added to the suppression pool takes 25.6 hours

4. Calculate probability of not recovering power or decay heat removal pump.

The probability of not recovering offsite power in 25.6 hours is Sum (25.6) No Obtuse Lines / Sum (0) No Obtuse Lines = 0.2359E-03 / 0.566E-01 = 4.2E-03 (Ref 4, p.A-237)

The probability of not recovering a diesel generator in 25.6 hours is 1.7E-01 (Ref 4, Table C.2-4)

The probability of not recovering decay heat removal pump in 25.6 hours is 5.4E-01

Probability = EXP (-(25.6 – 12) hrs / 22.3 hrs)

EXP = e raised to power in ( )
Planning time = 12 hours (Ref 4)
Mean time to repair a pump is 22.3 hours (Ref 33, Table 11).

5. Calculate revised PRA using the following inputs:

Probability that Offsite power not recovered in 25.6 hour.
NR26 = 4.2E-03

Probability of Failure to recover diesel generator at 25.6 hours
DGR26 = 1.7E-01

Probability of Failure to repair a pump used for decay heat removal at 25.6 hours
DHRpmpR = 5.4E-01

Table E-1
PRA Results with Containment Failure at 155 psia

Summary of Results - with normal maintenance and failure of containment at 155 psia

Plant Status	Frequency per 15 months	Frequency per 12 months
Initiating Event	2.43E+00	1.94E+0
CD-UO-COK	1.66E-07	1.33E-7
CD-OH-COK	1.01E-07	8.10E-8
Total CD	2.67E-07	2.14E-7
CD-HPVF	7.97E-10	6.37E-10
CD-LPVF-COK	3.72E-10	2.97E-10
Total Vessel Failure	1.17E-09	9.35E-10
CD-UO-ECF	2.79E-11	2.23E-11
CM-VF-COTF	2.15E-10	1.72E-10
LERF	2.43E-10	1.95E-10
CM-VOK-COPF	8.40E-11	6.72E-11
CM-VF-COPF	5.20E-11	4.16E-11
Late Cont. Failure	1.36E-10	1.09E-10
COPF Prior to Core Damage		
COPF	4.64E-07	3.71E-7
50% of COPF	2.32E-07	1.85E-7
Add Total CD to 50% COPF to account for CD after Containment Failure		3.99E-7

Note: The CDF for 155 psia (3.99E-07 / year) is slightly different than the CDF used in Section 4.0 (3.74E-07 / year). This is because there are some other mitigating measures such as mass addition to suppression pool that are not included in concealed corrosion analysis. The analysis is performed using the same inputs except as noted. The delta CDF between 155 psia (3.99E-07 / year) and 60 psia (4.49E-07 / year) is the critical value and the results are appropriate.

Time to reach 62 psia

The containment with corrosion fails at 62 psia. The elevated pressure is due to decay heat from the core and is the result of water vapor. The saturated temperature for 62 psia is 295 F. The bulk temperature of the suppression pool is 287 F because 8 F difference accounts for temperature stratification in the suppression pool.

1. Calculate the mass of steam added to the suppression pool as follows (Ref. 32, p.157):

Mass of steam added (lbm) =

$$\frac{\text{Mass of suppression pool (initial) x (final enthalpy of water – initial enthalpy of water)}}{\text{(final enthalpy of steam – final enthalpy of water)}}$$

Mass of suppression pool (initial) = 7,940,000 lbm
(This represents 7,600,000 lbm initially plus 340,000 lbm from the initial blowdown of the RPV to the suppression pool. This blowdown is included to account for a rapid depressurization associated with the HCTL.)

Final enthalpy of water = 256 BTU/lbm
Initial enthalpy of water = 106 BTU/lbm
(The suppression pool water temperature after the initial blowdown is 138 F)
Final enthalpy of steam = 1178 BTU/lbm

Mass of steam added (lbm) =

$$\frac{7{,}940{,}000 \text{ lbm x } (256 - 106) \text{ BTU/lbm}}{(1178\text{-}256) \text{ BTU/lbm}} = 1.29\text{E+06 lbm}$$

2. Calculate the energy of steam added to the suppression pool as follows:

Total energy added to pool (MW-sec) =

Mass of steam added x (Final enthalpy of steam – ((Final enthalpy of water + Initial enthalpy of water) x 0.5)) x 0.00106 MW-sec/BTU

Mass of steam added = 1.29E+06 lbm

Final enthalpy of steam = 1178 BTU/lbm
Final enthalpy of water = 256 BTU/lbm
Initial enthalpy of water = 106 BTU/lbm

Total energy added to pool (MW-sec) =

= 1.29E+06 lbm x (1187 – ((256 + 106) x 0.5)) BTU/lbm x 0.00106 MW-sec/BTU
= 1.37E+06 MW-sec

3. Time to 62 psia is calculated based on Eric Haskin Tabular Decay Heat (Ref 32, p.351)

Cumulative power of 1.37E+06 MW-sec added to the suppression pool takes 14.2 hours

4. Calculate probability of not recovering power or decay heat removal pump.

The probability of not recovering offsite power in 14.2 hours is Sum (14.2) No Obtuse Lines / Sum (0) No Obtuse Lines = 0.8232E-03 / 0.566E-01 = 1.5E-02 (Ref 4, p.A-237)

The probability of not recovering a diesel generator in 14.2 hours is 2.2E-01 (Ref 4, Table C.2-4)

The probability of not recovering decay heat removal pump in 14.2 hours is 9.1E-01

Probability = EXP (-(14.2 – 12) hrs / 22.3 hrs)

EXP = e raised to power in ( )
Planning time = 12 hours (Ref 4)
Mean time to repair a pump is 22.3 hours (Ref 33, Table 11).

5. Calculate revised PRA using the following inputs:

Probability that Offsite power not recovered in 14.2 hour.
NR26 = 1.5E-02

Probability of Failure to recover diesel generator at 14.2 hours
DGR26 = 2.2E-01

Probability of Failure to repair a pump used for decay heat removal at 14.2 hours
DHRpmpR = 9.1E-01

Table E-2
PRA Results with Containment Failure at 62 psia

Plant Status	Frequency per 15 months	Frequency per 12 months
Initiating Event	2.43E+00	1.94E+0
CD-UO-COK	1.66E-07	1.33E-7
CD-OH-COK	1.01E-07	8.10E-8
Total CD	2.67E-07	2.14E-7
CD-HPVF	7.97E-10	6.37E-10
CD-LPVF-COK	3.72E-10	2.97E-10
Total Vessel Failure	1.17E-09	9.35E-10
CD-UO-ECF	2.79E-11	2.23E-11
CM-VF-COTF	2.15E-10	1.72E-10
LERF	2.43E-10	1.95E-10
CM-VOK-COPF	3.29E-10	2.63E-10
CM-VF-COPF	1.50E-10	1.20E-10
Late Cont. Failure	4.79E-10	3.83E-10
COPF Prior to Core Damage		
COPF	5.87E-07	4.69E-7
50% of COPF	2.93E-07	2.35E-7
Add Total CD to 50% COPF to account for CD after Containment Failure		4.49E-7

Determine change in Class 7 frequency

Class 7 accidents are almost entirely composed of COPF sequences. Use 50% of COPF because not all containment failures occur with core damage.

From Table E-1, the Class 7 frequency with containment failure at 155 psia at 25.6 hours is 1.856E-07 / year.

From Table E-2, the Class 7 frequency with containment failure at 62 psia at 14.2 hours is 2.347E-07 / year.

Failure rate of 100% degradation of containment liner is based on 2 events among 70 plants in 5 years. The 2 events are Brunswick 2 and North Anna 2 (27,28). The 70 plants are based on industry data base of steel lined concrete containments (26). The 5 years is based on changes to 10 CFR50.55a that require periodic visual inspections of containment surfaces since September 1996 (29).

Failure rate = 2 events / (70 plants x 5 years) = 0.005714 / year

The revised Class 7 frequency =

((1 – EXP (- failure rate x time)) x Class 7 frequency at 62 psia) +
(EXP (- failure rate x time) x Class 7 frequency at 155 psia)

The results are as follows:

Time	Class 7 frequency	Delta from Class 7 frequency at 155 psia
3 yr	1.864E-07	8.29E-10
10 yr	1.883E-07	27.1E-10
15 yr	1.896E-07	40.1E-10

Step 3 - Calculate risk for 3 year concealed corrosion.

Table E-3 is derived the same way as Table 3 in Section 4.0 of the main calculation. The derivation for the Table is explained in Section 4.0.

Based on Table E-2, the CDF is changed to 4.49E-07, the LERF is changed to 1.95E-10, and Late Containment Failure is changed to 3.83E-10. The concealed corrosion affects only accident sequences that are part of Class 7 because a liner failure may result in loss of containment that is grouped as a Severe Accident. For Class 7, Tables E-3, E-4 and E-5 carries extra decimal places to show the small differences as the risk analysis progresses.

Also, the probability of Class 7 is increased by the following factor:

8.29E-10 / year is the increased frequency that a corrosion failure will occur in 3 years and cause containment to fail at 62 psia.

Table E-3
Mean Consequence Measures for 3-Year Concealed Corrosion Interval

Class	Description	EPRI analysis	Probability (P) Frequency (/12 month)	Consequence (C) Person-Rem to 50 miles	Risk (PxC) Person-rem/yr	Basis Frequency
1	No Containment Failure - use La	relevant	1.54E-07	3.29E+05	5.07E-02	Core Damage Frequency minus frequency of other classes. CDF 4.49E-7
2	Large containment isolation failure - use 35 La	Random failures to close - Type A not relevant	6.42E-10	4.38E+05	2.81E-04	1.43E-3 times CDF - method based on failure of Containment Isolation of pentrations > 1 inch
3a	Small isolation failure - use 10La	relevant	2.87E-08	4.41E+05	1.27E-02	0.064 times CDF - based on NUREG ILRT results of 4 small failures out of 144 tests - 95th percentile of Chi squared distribution
3b	Large isolation failure - use 35La	relevant	9.43E-09	4.38E+05	4.13E-03	0.021 times CDF - IP3 method based on NUREG ILRT results of 0 large failures out of 144 tests - 95th percentile of Chi squared distribution
4	Small isolation failure (Type B penetration)	Based on Type B frequency - not relevant	0	0	0.0	
5	Small isolation failure (Type C penetration)	Based on Type C frequency - not relevant	0	0	0.0	
6	Containment isolation failures (dependent failures, personnel error) - use 35 La	Based on ISI/IST program - Type A test does not affect	1.03E-09	4.38E+05	4.52E-04	2.3E-3 times CDF - EC-RISK-1063
7	Severe Accident induced failure - use 100 La	Type A test does not affect - not relevant	2.362E-07	6.27E+06	1.48E+00	SSES PRA results for LERF, Late Containment Failure, and 50% of Containment Over Pressure Failure (prior to core damage) plus 8.29E-10 increase for concealed corrosion
8	Secondary Containment bypassed - use 100 La	Type A test does not affect - not relevant	1.90E-08	4.24E+06	8.05E-02	ISLOCA plus Containment Bypass
Core Damage			4.49E-07		1.6297E+00	

Step 4 - Calculate risk for 10 year Concealed Corrosion Interval

The probability of Class 7 is increased by the following factor:

27.1E-10 / year is the increased frequency that a corrosion failure will occur in 10 years and cause containment to fail at 62 psia.

Table E-4
Mean Consequence Measures for 10-Year Concealed Corrosion Interval

Class	Description	EPRI analysis	Probability (P) Frequency (/12 month)	Consequence (C) Person-Rem to 50 miles	Risk (P x C) Person-rem/yr	Basis Frequency
1	No Containment Failure - use La	relevant	1.52E-07	3.29E+05	5.00E-02	Same as 3 year - Core Damage Frequency minus frequency of other classes
2	Large containment isolation failure - use 35 La	Random failures to close - Type A not relevant	6.42E-10	4.38E+05	2.81E-04	Same as 3 year
3a	Small isolation failure - use 10 La	relevant	2.87E-08	4.41E+05	1.27E-02	Same as 3 year
3b	Large isolation failure - use 35La	relevant	9.43E-09	4.38E+05	4.13E-03	Same as 3 year
4	Small isolation failure (Type B penetration)	Based on Type B frequency - not relevant	0	0	0.0	
5	Small isolation failure (Type C penetration)	Based on Type C frequency - not relevant	0	0	0.0	
6	Containment isolation failures (dependent failures, personnel error) - use 35 La	Based on TSI/IST program - Type A test does not affect	1.03E-09	4.38E+05	4.52E-04	Same as 3 year
7	Severe Accident induced failure - use 100 La	Type A test does not affect - not relevant	2.381E-07	6.27E+06	1.49E+00	Same as 3 year except 27.1E-10 added for concealed corrosion
8	Secondary Containment bypassed - use 100 La	Type A test does not affect - not relevant	1.90E-08	4.24E+06	8.05E-02	Same as 3 year
Core Damage			4.49E-07		1.6409E+00	

The percent risk increase ($\Delta\%Risk_{10}$) due to a ten-year Concealed Corrosion over the baseline case is as follows:

Delta %Risk = ((Total-10 – Total-base) / Total-base) * 100

Where:

Total-base = total person-rem / year for baseline interval = 1.6297 person-rem / year
Total-10 = total person-rem / year for 10-year interval = 1.6409 person-rem / year

Delta %Risk = ((1.6409 – 1.6297) / 1.6297) * 100
= 0.69%

Therefore, the increase in risk contribution because of concealed corrosion from 3 years to 10 years is 0.69%

Step 5 - Risk Impact due to 15 year Concealed Corrosion Interval

The probability of Class 7 is increased by the following factor:

40.1E-10 / year is the increased frequency that a corrosion failure will occur in 15 years and cause containment to fail at 62 psia.

Table E-5
Mean Consequence Measures for 15-Year Concealed Corrosion Interval

Class	Description	EPRI analysis	Probability (P) Frequency (/12 month)	Consequence (C) Person-Rem to 50 miles	Risk (P x C) Person-rem/yr	Basis Frequency
1	No Containment Failure - use La	relevant	1.51E-07	3.29E+05	4.96E-02	Core Damage Frequency minus frequency of other classes
2	Large containment isolation failure - use 35 La	Random failures to close - Type A not relevant	6.42E-10	4.38E+05	2.81E-04	Same as 3 year
3a	Small isolation failure - use 10La	relevant	2.87E-08	4.41E+05	1.27E-02	Same as 3 year
3b	Large isolation failure - use 35La	relevant	9.43E-09	4.38E+05	4.13E-03	Same as 3 year
4	Small isolation failure (Type B penetration)	Based on Type B frequency - not relevant	0	0	0.0	
5	Small isolation failure (Type C penetration)	Based on Type C frequency - not relevant	0	0	0.0	
6	Containment isolation failures (dependent failures, personnel error) - use 35 La	Based on ISI/IST program - Type A test does not affect	1.03E-09	4.38E+05	4.52E-04	Same as 3 year
7	Severe Accident induced failure - use 100 La	Type A test does not affect - not relevant	2.394E-07	6.27E+06	1.50E+00	Same as 3 year except 40.1E-10 added for concealed corrosion
8	Secondary Containment bypassed - use 100 La	Type A test does not affect - not relevant	1.90E-08	4.24E+06	8.05E-02	Same as 3 year
Core Damage			4.49E-07		1.6486E+00	

The percent increase on the total integrated plant risk for these accident sequences is computed as follows.

Delta %Risk = ((Total-15 – Total-10) / Total-10) * 100

Where:

Total-15 = total person-rem / year for 15 year interval = 1.6486 person-rem / year
Total-10 = total person-rem / year for 10 year interval = 1.6409 person-rem / year

Delta %Risk = ((1.6486 – 1.6409) / 1.6409) * 100
= 0.47%

Therefore, the increase in risk contribution because of concealed corrosion from 10 years to 15 years is 0.47%

The percent risk increase ($\Delta\%Risk_{15}$) due to a fifteen-year concealed corrosion over the baseline case is as follows:

Delta %Risk = ((Total-15 – Total-base) / Total-base) * 100
Where:

Total-15 = total person-rem / year for 15 year interval = 1.6486person-rem / year
Total-base = total person-rem / year for 3 year interval = 1.6297 person-rem / year

Delta %Risk = ((1.6486 – 1.6297) / 1.6297) * 100
= 1.16%

Therefore, the increase in risk contribution because of concealed corrosion from 3 years to 15 years is 1.16%

Step 6 - Calculate change in LERF

The risk impact associated with concealed corrosion involves the potential that a core damage event that normally would result in only a small radioactive release from containment could in fact result in a large release due to failure to detect a pre-existing containment liner failure during the relaxation period. For this evaluation only Class 7 sequences have the potential to result in large releases if a pre-existing corrosion were present.

Reg. Guide 1.174 (3) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 (3) defines very small changes in risk as resulting in increases of core damage frequency (CDF) below 1E-06/yr and increases in LERF below 1E-07/yr. Since the concealed corrosion does not impact CDF, the relevant metric is LERF. Calculating the increase in LERF requires determining the impact of the concealed corrosion on the leakage probability.

The analysis described that the containment will not fail for 14 hours due to concealed corrosion. For this analysis, 14 hours is considered early enough to lead to an early release. The LERF is considered to be the sum of class 3b and class 7 sequences. The following table is based on results in Table E-3, E-4, and E-5 from above.

LERF Results (per year)

	Base line - 3 year interval	10 year interval	15 year interval
LERF (class 3b)	9.43E-09	9.43E-09	9.43E-09
LERF (class 7)	2.36E-07	2.38E-07	2.39E-07
Total LERF	2.46E-07	2.48E-07	2.49E-07
Increase over 3 year		1.88E-09	3.18E-09
Increase over 10 year			1.30E-09

E. Results

The risk increase in extending ILRT from 10 years to 15 years on the total integrated plant risk due to corrosion of the containment liner from the concealed surface is 0.47%. The increase in LERF is 1.30E-09 / year. Both of these increases are not significant.

Total Integrated Risk Comparison with Precedent Approvals

Several utilities have received approval to extend the ILRT to 15 years as is analyzed in this calculation. The below table summarizes the risk analyses results of those previously approved and compares those results with the PPL results.

Utility	Total Integrated Risk Increase (%) (1/10 to 1/15)	Total Integrated Risk Increase (%) (3/10 to 1/15)	Comments[1]
Results WITHOUT consideration of concealed corrosion[2].			
Peach Bottom	.04	.12	NRC SER indicates these values are "small"
Crystal River	.045	.14	NRC SER indicates these values are "small"
Indian Point 3	.048	.43	NRC SER indicates these values are "small"
PPL	.02	.05	
Results WITH consideration of concealed corrosion.			
PPL	.47	1.16	

Observations:

1. The PPL risk impact, when concealed corrosion is not considered, is bounded by the previously approved precedents. It therefore is concluded that the risk impact of this change when concealed corrosion is not considered is not significant and considered "small", as was concluded for the precedents.
2. The 1/10 to 1/15 risk increase value is the relevant risk value since it directly reflects the change being requested. That is, the risk increase from performing one Type A test every 10 years (current PPL requirement) compared to performing one test

[1] In each of the SER's, NRC compared the results to those contained in NUREG-1493. This NUREG states that the typical increase in risk for extending the ILRT from 3 tests in 10 years to 1 test in 10 years will be 0.02% to 0.14%. It describes this increase as "imperceptible".

[2] All the analyses were done in accordance with EPRI TR-104285. The ERPI TR-10485 does not include an assumption to fail containment at 62 psia for Class 7 sequences that was requested by NRC of PPL to account for the potential for concealed corrosion. As a result, the analysis in this Appendix E is more conservative than NUREG-1493, and predecessor submittals.

every 15 years (proposed requirement) is the relevant metric. The other risk increase metrics serve only to put the values in context of what was analyzed and published in the NUREG-1493, though they do not directly relate.

3. If the precedents had analyzed concealed corrosion, the risk values would be higher.
4. The PPL 1/10 to 1/15 risk increase value (.46%) is essentially equal to the total integrated risk value (.43%) found acceptable for Indian Point 3. Thus it can be concluded that the relevant risk increase (1/10 to 1/15) even when concealed corrosion is considered, is "small" and thus is acceptable.
5. PPL concludes that the 1.16 % risk increase calculated for the 3/10 to 1/15 case is small.

Attachment 1 to PLA-5424

EC-RISK-1081 Appendix E

Containment Fragility Calculation

Concurrently, the frequency of SR 3.6.1.1.2 is also deferred. This SR is performed to determine that the drywell-to-suppression chamber leakage is within limits. The SR is performed as part of the Type A test evolution and thus is required whenever the Type A test is performed.

The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment is provided below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   The frequency of Type A testing does not change the probability of an event that results in core damage or vessel failure. Primary containment is the engineered feature that contains the energy and fission products from evaluated events. The SSES IPE documents events that lead to containment failure. The frequency of events that lead to containment failure does not change because it is not a function of the Type A test interval. Containment failure is a function of loss of safety systems that shutdown the reactor, provide adequate core cooling, provide decay heat removal, and loss of drywell sprays.

   Similarly, the frequency of the SR 3.6.1.1.2 bypass test does not change the probability of an event that results in core damage or vessel failure since they are not a function of the bypass test.

   The consequences of the evaluated accidents are the amount of radioactivity that is released to secondary containment and subsequently to the public. Normally, extending a test interval increases the probability that a Structure, System, or Component will fail. However, NUREG-1493, Performance-Based Containment Leak-Test Program, states that calculated risk in BWR's is very insensitive to the assumed leakage rates. The remaining testing and inspection programs provide the same coverage as these tests, and will maintain leakage at appropriately low levels. Any leakage problems will be identified and repairs will be made. Additionally, the containment is continuously monitored during power operation. Anomalies are investigated and resolved. Thus there is a high confidence that integrity will be maintained independent of the Type A test and SR 3.6.1.1.2 bypass test frequency.

   Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously analyzed?

   Primary containment is designed to contain energy and fission products during and after an event. The SSES IPE identifies events that lead to containment failure. The proposed revision to the Type A and SR 3.6.1.1.2 test interval does not change this list of events. There are no physical changes being made to the plant and there are no changes to the operation of the plant that could introduce a new failure mode creating an accident or affecting mitigation of an accident.

   Therefore, this proposed amendment does not involve a possibility of a new or different kind of accident from any previously analyzed.

3. Does the proposed change involve a significant reduction in a margin of safety?

   The proposed one time extension to the Type A test frequency and the frequency of SR 3.6.1.1.2 from 10 to 15 years does not involve a significant reduction in margin of safety.

   The tests are performed to ensure the degree of reactor containment structural integrity and leak-tightness considered in the plant safety analysis is maintained. These proposed changes do not affect the degree of leak-tightness nor structural integrity of the containment. These proposed changes only affect the frequency by which the tests are performed. The test acceptance criteria are not affected.

   The proposed TS changes do not involve a change in the manner in which any plant system is operated or controlled.

   The proposed TS changes do not affect the availability of equipment associated with containment integrity that is assumed to operate in the plant safety analysis.

   The NUREG-1493 generic study of the effects of extending containment leakage testing found that a 20-year interval in Type A leakage testing resulted in an imperceptible increase in risk to the public. PPL analyses determined the total integrated risk and LERF increase is not significant. NUREG-1493 found that, generically, the design containment leakage rate contributes a very small amount of individual risk and would have minimal affect since most potential leakage paths are detected by Type B and Type C testing. Type B and Type C testing combined with visual inspection programs will maintain containment leakage at appropriately low levels.

The vacuum breaker leakage test (SR 3.6.1.1.3) and stringent acceptance criteria, combined with the negligible non-vacuum breaker leakage area and thorough periodic visual inspection, provide an equivalent level of assurance as the SR 3.6.1.1.2 bypass test. PPL analyses determined the total integrated risk and LERF increase is not significant.

The combination of the factors described above ensures that the proposed changes do not represent a significant reduction on margin of safety.

Based upon the above, the proposed amendment does not involve a significant hazards consideration.