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DEC 0 3 2001

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SUQUEHANNA STEAM ELECTRIC STATION
SUPPLEMENTAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING GENERIC LETTER 96-06 DATED JULY 26, 2001
PLA-5400

Docket Nos. 50-387 and 50-388

References: 1) *PLA-5093, R. G. Byram (PPL) to USNRC, "Generic Letter 96-06 Risk Assessment," dated August 3, 1999.*
2) *USNRC to R. G. Byram (PPL), "Request for Additional Information Regarding Supplemental Response to Generic Letter 96-06 (TAC Nos. M96875 and M96876)", dated July 26, 2001.*
3) *PLA-5352, R. G. Byram (PPL) to USNRC, "Response to Request for Additional Information Regarding Supplemental Response to Generic Letter 96-06 Dated July 26, 2001," dated September 5, 2001.*

On July 26, 2001, the NRC staff transmitted a request for additional information regarding the PPL Susquehanna LLC (PPL) risk assessment generated in response to Generic Letter 96-06, *"Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions."* PPL's response, dated September 5, 2001, documented the following commitment in response to NRC Question 8:

"PLA-5352 - 2 A quantitative analysis is being pursued with the valve vendor in order to determine if the subject valves are capable of accommodating the predicted post-LOCA pressures. The results of the analysis will be provided to the NRC by November 30, 2001."

The attachment to this letter documents the results of the analysis in question, and is provided in the form of an updated response to Question 8.

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

Very truly yours,

R. G. Byram

Attachment

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

A072

Revised Response to NRC Question 8

NRC Question 8:

Provide the results of piping and valve analysis based on the criteria contained in the American Society of Mechanical Engineers Code, Section III, Appendix F. For each component, provide a summary of the maximum faulted pressure, design load combination, calculated stress for design load combination including faulted pressure, and allowable stress based on the criteria contained in Appendix F. Also, you should include a reference to the specific provisions of Appendix F used as a basis in calculating the allowable stress (e.g., F-1331, F-1430, F-1420).

PPL Response:

The eleven containment piping penetrations (per unit) identified as being susceptible to thermally induced overpressurization have been evaluated for their pressure retention capability. The process piping located between the containment isolation valves associated with each penetration was evaluated using the criteria provided in the ASME Boiler & Pressure Vessel Code, Section III, Appendix F. Paragraph F-1430 has been used as a basis for calculating the allowable stresses. The results of the evaluation are provided here.

F-1430(a) states that the internal pressure shall not exceed 200% of the Design Pressure calculated in accordance with Eq.(2) of NB-3641.1. An allowable pressure for each piping penetration was determined using Eq.(2). The pressure limit is based on nominal wall thickness with a corrosion allowance. Table 1 below provides the allowable pressure for each penetration along with maximum post-LOCA temperatures and pressures.

The results demonstrate that the predicted maximum pressures for all of the lines are within the allowable pressure limits.

F-1430(b) states that Eq.(9) of NB-3652 shall be satisfied using a stress limit of the lesser of $3.0S_m$ or $2S_Y$. The hoop stress portion of Eq.(9) was determined using the maximum post-LOCA pressure associated with each line. The maximum Faulted stress for each line was extracted from the existing piping calculations and used for the mechanical stress portion of Eq.(9). These existing stresses are based on design basis Faulted load combinations that include pressure, deadweight, seismic and hydrodynamic loadings. Table 2 provides a tabulation of the Eq.(9) stresses and the allowables used.

The results demonstrate that all of the piping stresses are within allowable Appendix F limits.

A qualitative assessment of each penetration was performed in order to assess the susceptibility of the subject valves to an overpressurization failure. Based on a review of the isolation valves associated with the eleven penetrations, nine contain inboard and outboard isolation valves of a different type and design. It has been concluded that because of the distinct design characteristics of these valves, the likelihood of simultaneous failure of the inboard and outboard isolation valves is negligible.

The remaining two penetrations (X-23 and X-24) contain isolation valves that are of the same design (flex-wedge gate valves). A quantitative analysis has been performed by the valve vendor in order to determine if the subject valves are capable of accommodating the predicted post-LOCA pressures. This analysis employed allowable stress criterion based on ASME Section III, Appendix I. The analysis concludes that the body-bonnet flange would leak prior to reaching 900 psi. This leakage would serve to release any build-up of pressure and thus would preclude overpressurization failure. It is therefore, concluded that gross failure of the valves is not expected.

In addition, the limiting pressure associated with the structural capability of the subject valves is 3566 psi which bounds the maximum post-LOCA pressure of 2280 psi (X-23 penetration) and 2420 psi (X-24 penetration).

Table 1. Pressure Limits In Accordance With F-1430(a)

Penetration	D_{pipe}	P_{max} (psi)	T_{max} (F)	$P_{allowable}$ (psi)	Comments
X-85A	3”	3010	118	5143	$P_{max} < P_{allowable}$
X-85B	3”	2810	120	5143	$P_{max} < P_{allowable}$
X-86A	3”	4570	139	5143	$P_{max} < P_{allowable}$
X-86B	3”	4400	142	5143	$P_{max} < P_{allowable}$
X-23	4”	2280	133	4360	$P_{max} < P_{allowable}$
X-24	4”	2420	135	4360	$P_{max} < P_{allowable}$
X-17	6”	4600	160	5473	$P_{max} < P_{allowable}$
X-54	8”	2570	114	3059	$P_{max} < P_{allowable}$
X-53	8”	2970	117	3059	$P_{max} < P_{allowable}$
X-56	8”	2570	114	3059	$P_{max} < P_{allowable}$
X-55	8”	3030	114	3059	$P_{max} < P_{allowable}$

Table 2. Stress Limits In Accordance With F-1430(b)

Penetration	D_{pipe}	Hoop Stress (psi)	Mech Stress (psi)	Total Stress (psi)	3.0 S_m or 2.0 S_Y (psi)
X-85A	3”	12193	16399	28592	60000
X-85B	3”	11383	16038	27421	60000
X-86A	3”	18513	10324	28837	60000
X-86B	3”	17824	9122	26946	60000
X-23	4”	10823	20451	31274	60000
X-24	4”	11487	10166	21653	60000
X-17	6”	17636	21853	39489	50200
X-54	8”	17210	13668	30878	60000
X-53	8”	19888	13576	33464	60000
X-56	8”	17210	10255	27465	60000
X-55	8”	20290	10118	30408	60000

November 8, 2001

SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO COMMITMENT PLA-5352-2
RESULTS OF VALVE VENDOR ANALYSIS OF
SUBJECT VALVES CAPABILITY TO ACCOMADATE
THE PREDICTED POST-LOCA PRESSURES

REFERENCES:

1) G. T. Jones to USNRC, "Response to Request For Additional Information Regarding Supplemental Response to GL96-06", 09/05/2001 (PLA-5352)
2) Flowserve Log No. TR01.094 Rev.-, Pressure Capability Analysis For PPL Susquehanna Steam Electric Station, 4-Inch Class 150 Carbon Steel Flex Wedge Gate Valve with Limitorque Motor Actuator

This letter provides the PPL Susquehanna, LLC response to commitment PLA-5352-2, as documented in PLA-5352 (Reference 1). The commitment states that a quantitative analysis will be pursued with the valve vendor in order to determine if the subject valves are capable of accommodating the predicted post-LOCA pressures.

The results from the valve vendor's analysis ~~(Reference 2)~~ indicate that the maximum pressure that the subject valves can withstand is 603 psi, ~~see table "Maximum Allowable Pressure"~~ below. This limit is based on the body-bonnet flange. The failure would not present itself in material yielding, but it would act as a pressure relief due to inadequate gasket preload. The exact pressure where flange leakage would occur is uncertain because of uncertainty in sealing properties of the gasket. The vendor states that based on experience, they believe leakage is more likely to occur before the limiting pressure is reached. However, it is their judgement that an absolute upper pressure limit is conservatively predicted to be 900 psi.

In addition, the limiting pressure associated with the structural capability of the subject valves is 3566 psi, see table "Maximum Allowable Pressure" below, and bounds the maximum post-LOCA pressure of 2280 psi (X-23 penetration) and 2420 psi (X-24 penetration), as documented in PLA-5352 (Reference 1).

Maximum Allowable Pressure	
Component	Limiting Pressure (psi)
Body-Bonnet Flange & Bolting	603
Gland Flange	5900
Gland Bolt & Cross Pin	9449
Disc	3566

Note that the vendor analysis employed allowable stress criterion based on the design rules of the ASME Boiler & Pressure Vessel Code, Section III.

Mike
Crowthers
GENAC01

NUCLEAR DEPARTMENT
ACTION REQUIRED

ISSUE DATE **10/19/01**

ACTION GROUP **MNTT/MHR**
(LEAD FUM FOR ACTION ITEM)

DUE DATE **10/30/01**

CYCLE ______

TRACKING # **NIMS #363320**

REFERENCE DOCUMENT # **PLA-5352-2**

☐ CONDITION REPORT

☐ EMERGENCY PLAN

☐ AUDIT RECOMMENDATION

☐ SURVEILLANCE OBSERVATION

☐ ALARA ______

☐ PORC ACTION ITEM

☐ OTHER ______

ACTION ITEM:

☐ ASSESSMENT
- ___ ISES
- ___ ASSESSMENT GROUP
- ___ OTHER ______

☒ NRC
- ___ VIOLATION
- ___ UNRESOLVED ITEM
- ✓ OTHER COMMITMENT
- ___ OTHER ACTION

☐ OTHER REGULATORY

☐ INDUSTRY

☐ INPO
- ___ EVALUATION
- ___ SOER

SUMMARY OF REQUIRED ACTION: Provide results of the quantitative analysis from the valve vendor to determine if the subject valves are capable of accommodating the predicted post-LOCA pressures, to the Licensing Group. NRC Commitment from attached PLA-5352, "Response To Request For Additional Information Regarding Supplemental Response To Generic Letter 96-06 Dated July 26, 2001".

LEAD GROUP / CONTACT: **M. H. CROWTHERS**

ACTION TAKEN:

See attached letter dated 11/8/2001, Response to Commitment PLA-5352-2

[signature] 11/8/2001
PREPARER SIGN / DATE

[signature] 11/8/01
Supervisor ~~ACTION FUM SIGN~~ / DATE
(per telecon with Mike Crowthers)

RETURN TO: Donna M. Davidson, LIC, NUCSA4

Formerly NDAP-QA-0702-12

bc:	M. R. Anthony	NUCSB3	w/o	
	K. G. Browning	GENA63	w/o	
	F. G. Butler	GENA63	w/o	
	M. H. Crowthers	GENA63	w/a	
	T. L. Harpster	GENA61	w/o	
	G. T. Jones	GENA61	w/o	
	R. D. Kichline	GENA61	w/o	
	C. Kukielka	GENA63	w/o	
	G. G. Maertz	NUCSA2	w/o	
	M. R. Mjaatvedt	GENA63	w/o	
	R. D. Pagodin	GENA63	w/o	
	M. H. Rose	GENA62	w/o	
	R. R. Sgarro	GENA61	w/o	
	J. D. Shaw	NUCSB2	w/o	
	B. L. Shriver	NUCSB3	w/o	
	M. W. Simpson	GENA12	w/o	
	Licensing Files	GENA61	w/a	(w/orig. attached NDAP-QA-0729-1)
	Nuclear Records Attn: P. Brown	GENA62	w/a	(w/attached NDAP-QA-0729-1)

George T. Jones
Vice President
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SUQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING SUPPLEMENTAL RESPONSE TO GENERIC LETTER 96-06 DATED JULY 26, 2001
PLA- 5352

Docket Nos. 50-387 and 50-388

Reference:
1) R. G. Byram to USNRC, "Generic Letter 96-06 Risk Assessment," dated August 3, 1999 (PLA-5093)
2) USNRC to R. G. Byram "Request for Additional Information Regarding Supplemental Response to Generic Letter 96-06 (TAC Nos. M96875 and M96876), dated July 26, 2001

This letter provides the PPL Susquehanna, LLC (PPL) response to your request for additional information regarding Generic Letter 96-06 (Reference 2) serving to supplement the PPL risk-based assessment of the potential for thermally-induced over-pressurization of containment piping penetrations. The following commitments are made herein:

PLA-5352 - 1 The analysis results assume the addition of insulation to two sections of piping located in the drywell of each unit. These two sections will be insulated during the next two refueling outages. These are scheduled to occur in the Spring 2002 for Unit 1 and Spring 2003 for Unit 2.

PLA-5352 - 2 A quantitative analysis is being pursued with the valve vendor in order to determine if the subject valves are capable of accommodating the predicted post-LOCA pressures. The results of the analysis will be provided to the NRC by November 30, 2001.

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

Very truly yours,

G. T. Jones

Attachments

copy: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. Schaaf, NRC Project Manager

RAI Responses

NRC Question 1

Provide further justification for P[3] - containment heating causes heating and expansion of the water trapped between the isolation valves over-pressurizing the pipe until rupture. At a minimum, please, address additional failure modes such as station blackout and human reliability.

PPL Response

It was assumed in the Reference 1 analysis that penetration rupture is a certainty in the absence of drywell sprays. This assumption was made since detailed pipe rupture analysis had not yet been performed and since it was conservative to assume pipe rupture occurs.

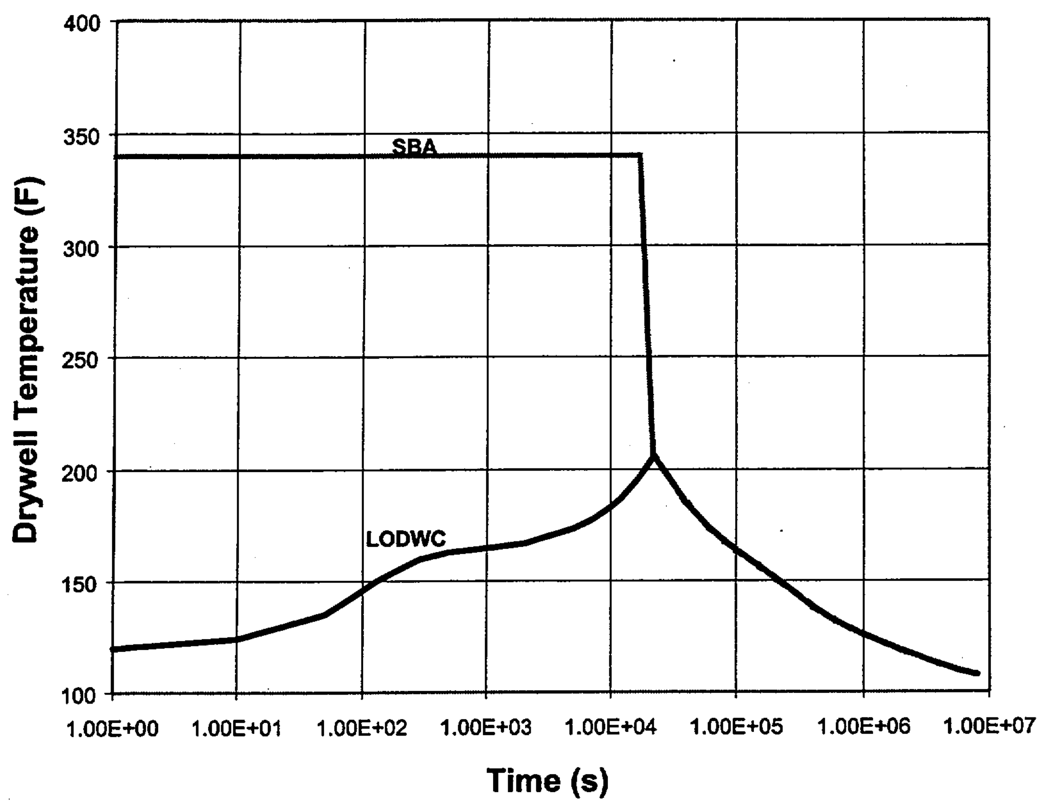
Subsequently, PPL performed detailed heat transfer analyses to evaluate the impact containment heating has on each susceptible penetration's internal pressure. The response to question 7 describes the heat transfer analysis in detail.

Figure 1 shows the temperature profile associated with the small break LOCA. This temperature profile bounds all other temperature profiles including large break LOCA and Loss of Drywell Cooling. The Loss of Drywell Cooling event provides the bounding temperature for all non-LOCA MSIV closure events such as Station Blackout. The analysis results described in the responses to questions 7 and 8 show that no penetration experiences an increase in internal pressure sufficient to cause a pipe failure for the bounding small break LOCA temperature profile.

The analysis results assume the addition of insulation to two sections of piping located in the drywell of each unit. These two sections will be insulated as the analysis assumes during the next two refueling outages. These are scheduled to occur in the Spring 2002 for Unit 1 and Spring of 2003 for Unit 2.

The only operator action assumed in the analysis is the initiation of shutdown cooling six hours after the small break LOCA. Emergency operating procedures direct the operators to initiate shutdown cooling. It is expected that the 98 psig interlock will clear in 3-4 hours after the event occurs. Thus the 6-hour assumption bounds the expected time for initiation of shutdown cooling. Operation of shutdown cooling prior to 6 hours will shorten the time in which the drywell is at the peak temperature shown in Figure 1.

Figure 1 Drywell Temperature Profiles for Small Break LOCA Accident (SBA) and Loss of Drywell Cooling (LODWC).



NRC Question 2:

Page 16 of the submittal states, "Penetration failure is a concern when a large radioactive source term is available for release in the drywell." Please quantify the probability of a large radioactive source term in the drywell at the time of penetration failure.

PPL Response:

The presence of a large radioactive source term will exist in the drywell given core damage. The Core Damage Frequency (CDF) at Susquehanna is estimated to be 5E-7 as reported in the NRC SER on the Susquehanna Individual Plant Examination, (Letter from Victor Nerses to Robert Byram Review of Susquehanna IPE August 11, 1998). As described in the response to questions 7 and 8, penetration failure is not deemed credible.

NRC Question 3:

During the resolution of Generic Safety Issue 150, "Over-pressurization of Containment Penetrations," the staff estimated a value of 0.1 for the probability that the penetration fails in a manner that results in a leakage path from the containment atmosphere to the environment. Although the staff believes that 0.1 is very conservative, barring further justification from PPL, the staff believes that a value of 0.1 is more appropriate than PPL's estimate of between 10^{-5} and 10^{-2} with a point estimate of $3x10^{-4}$. The staff's concern in supporting a less conservative value is based on PPL's application of Branch Technical Position MEB 3-1, failure to address the effects of non-uniform strain, failure to identify the more likely failure points given over-pressurization, and a lack of relevant data.

PPL Response:

The probability that a penetration fails in a manner that results in a leakage path from the containment atmosphere to the environment is identified as P[4] in Reference 1. Subsequent to the analysis submitted in Reference 1 and despite concluding that the internal pressure will not be sufficient to cause penetration failure as described in response 7, PPL and Structural Integrity Associates (SIA) performed specific failure analysis of each susceptible penetration.

These additional analyses, summarized below, conclude that even if the penetrations were to fail, it is highly unlikely that any of the identified penetrations will fail in a manner that will result in a leak path from the containment to the environment.

Penetration X-72B Drywell Sump Pump Discharge

As discussed in Reference 1, this penetration is not susceptible to overpressurization for a number of reasons.

1. The line only contains water when the pump is running to drain the sump, which is about 1% per year.
2. If an isolation signal occurs while the pump is operating, the pump trip precedes the signal to close the isolation valves. This design allows the piping to drain prior to isolation.
3. Both isolation valves are outside the drywell and are therefore not susceptible to heating.
4. The piping connecting to the drywell penetration is isolated by the leak tight containment isolation valve at the containment penetration and by four check valves at the sump. Leakage of any of the four check valves will relieve any possible pipe pressurization.
5. The containment isolation valve is welded directly to the containment penetration outside of the containment. Process piping is welded to the penetration inside of the containment. Should penetration failure occur it would occur at the process piping inside the containment.

Based on the above, this penetration is not susceptible to failure.

Penetrations X-53, X-54, X-55, X-56, X-86A & B RBCCW Piping to and from the Drywell Coolers and Recirculation Pumps

These penetrations are not expected to fail because the internal pressure of the pipe should remain below the yield strength of the pipe. However if containment heating did result in penetration failure, the failure would occur in the containment and would not propagate through the containment wall.

The outboard isolation valves are welded directly to the imbedded pipe sleeve, while the inboard isolation valve is welded to process piping. The process piping in turn is welded to the pipe sleeve. The pressure retaining capability of the pipe sleeves are 1.75 to 4.5 times greater than the process piping. The pressure retaining capability of the valves themselves is greater than the process piping. Therefore if a failure were to occur between the isolation valves it would fail at the process piping.

Furthermore, the isolation valves are of different design. The inboard valves are butterfly valves and the outboards are gate valves. Therefore, if the isolation valves are the weak link, it is incredible that valves of different design would fail at exactly the same

pressure. Thus a penetration failure that would result in a leak path from the primary containment to the environment is not credible for these penetrations.

Penetrations X-23 & X-24 RBCCW Piping to and from the Recirculation Pumps

These penetrations are similar to those discussed immediately above, with the exception that both the inboard and outboard isolation valves are of the same design. As in the case above, if failure occurs, it is expected to occur in the process piping located in the drywell and not the penetration. Thus a penetration failure that would result in a leak path from the primary containment to the environment is not credible.

Penetrations X-85A & X-85B RBCW Piping to and from the Recirculation Pumps

These penetrations have process piping between the penetration sleeve and both the inboard and outboard valves. The inboard valves are wafer style butterfly valves and the outboard valves are double disc gate valves. It is anticipated that the inboard butterfly valves will leak at the gasket and relieve pressure prior to process piping rupture. Calculations demonstrate that the butterfly valve flange will leak significantly (60 lb_m/hr) at 1850 psi which is more than a factor of 5 below the burst pressure of the process pipe. Thus a penetration failure that would result in a leak path from the primary containment to the environment is not credible.

Penetration X-17 RHR Head Spray Penetration

The head spray penetration consists of a flued head design. The outboard isolation valve is welded directly to the flued head while the process piping is between the inboard valve and the penetration sleeve. Therefore, if penetration were to fail, it would fail in the containment.

NRC Question 4:

Section 5.0 of the submittal discusses two mitigating measures that provide protection to primary containment integrity for the over-pressurization failure mode as well as other threats. The submittal goes on to say that these measures have been implemented in the plant's Emergency Operating Procedure (EOPs) via safety evaluations per 10 CFR 50.59. Please provide a copy and reference these safety evaluations in your submittal. What control exists to assure that these improvements to the EOPs will not be modified without the consideration of the issues raised in GL 96-06?

PPL Response:

Two operator actions designed to prevent containment penetration failure credited in Reference 1 are operation of the drywell sprays and not reestablishing drywell cooling given a verified LOCA. A discussion of each action follows.

Credit for drywell sprays is no longer necessary to ensure penetration integrity. More detailed heat transfer calculations (See response to questions 7 & 8) have been performed to evaluate the impact of containment heating on each penetration's internal pressure. As discussed in the response to questions 7 & 8, heating of the isolated penetration does not raise the penetration internal pressure to the point of failure assuming no credit for the drywell sprays. Initiation of drywell sprays is no longer credited as an action necessary for maintaining penetration integrity. Therefore compliance with GL 96-06 is no longer a reason for drywell spray operation and is not incorporated into the EOP bases or safety evaluations.

Re-initiation of drywell cooling is forbidden if the containment isolation is the result of a LOCA. The restriction is documented on page 5 of 21 under "High Drywell Temperature" in Attachment 1 safety evaluation for EO-000-103 Primary Containment Control. The restriction is captured in emergency procedure "Restoring Drywell Cooling with a LOCA Signal Present". Changes to procedures are controlled via quality related procedure "Nuclear Department Procedure Program". Additionally, emergency procedures are part of the PPL EOP program, which are controlled via the PPL quality related procedure "Symptom-Oriented EOP and EP-DS Program and Writer's Guide".

NRC Question 5:

Considering the safety importance of the drywell spray valves to open as described in Section 4.2.1 of the submittal, what monitoring program will be implemented to support the modeling assumptions of the drywell spray isolation valves? Have insights from the engineering evaluation in Section 4.2.1 been incorporated into the drywell spray isolation valves' maintenance program? Have these valves been classified as having high safety significance? If so, will possible future changes to their classification consider their role in the disposition of GL 96-06?

PPL Response:

Operation of the drywell sprays is no longer credited to prevent failure of the penetrations as discussed in responses to question 7 and 8. Therefore, they are no longer significant from a GL 96-06 perspective.

NRC Question 6:

Page 16 of the submittal states that, "Estimating the probability that the containment will reach a sustained temperature sufficient to rupture requires an evaluation of ... the containment temperature for a spectrum of accidents," and, "It is assumed that penetration failure will occur if cooling to the drywell is not restored." Please provide the evaluation of containment temperature for a spectrum of accidents described above. Have you quantified the impact of drywell sprays on containment temperature? If not, what is the basis for the assumption that the penetration will not fail give restoration of cooling to the drywell?

PPL Response:

The bounding temperature profile is provided in the response to question 1 and discussed in response to question 7. The small break LOCA bounds all other accidents from a containment temperature perspective. The containment temperature response given drywell sprays is not provided since drywell sprays are no longer credited in the analysis.

NRC Question 7:

For those penetrations that are susceptible to thermally-induced over-pressure, provide the maximum-calculated temperature and pressure for the piping run. Describe in detail the method used to calculate these pressure and temperature values. This should include a discussion of the heat transfer model, and the basis for the heat transfer coefficients used in the analysis. Discuss any source of uncertainty associated with the calculated pressure and temperature.

PPL Response:

Attachment 2 provides a description of the method and includes discussion of the heat transfer model and the basis for the heat transfer coefficients used in the analysis. The attachment also discusses method validation and the sources of uncertainty associated with the calculated pressure and temperature.

NRC Question 8:

Provide the results of piping and valve analysis based on the criteria contained in the American Society of Mechanical Engineers Code, Section III, Appendix F. For each component, provide a summary of the maximum faulted pressure, design load combination, calculated stress for design load combination including faulted pressure, and allowable stress based on the criteria contained in Appendix F. Also, you should include a reference to the specific provisions of Appendix F used as a basis in calculating the allowable stress (e.g., F-1331, F-1430, F-1420).

PPL Response:

The eleven containment piping penetrations (per unit) identified as being susceptible to thermally induced overpressurization have been evaluated for their pressure retention capability. The process piping located between the containment isolation valves associated with each penetration was evaluated using the criteria provided in the ASME Boiler & Pressure Vessel Code, Section III, Appendix F. Paragraph F-1430 has been used as a basis for calculating the allowable stresses. The results of the evaluation are provided here.

F-1430(a) states that the internal pressure shall not exceed 200% of the Design Pressure calculated in accordance with Eq.(2) of NB-3641.1. An allowable pressure for each piping penetration was determined using Eq.(2). The pressure limit is based on nominal wall thickness with a corrosion allowance. Table 1 below provides the allowable pressure for each penetration along with maximum post-LOCA temperatures and pressures.

The results demonstrate that the predicted maximum pressures for all of the lines are within pressure limits assuming nominal wall thickness values.

F-1430(b) states that Eq.(9) of NB-3652 shall be satisfied using a stress limit of the lesser of $3.0S_m$ or $2S_Y$. The hoop stress portion of Eq.(9) was determined using the maximum post-LOCA pressure associated with each line. The maximum Faulted stress for each line was extracted from the existing piping calculations and used for the mechanical stress portion of Eq.(9). These existing stresses are based on design basis Faulted load combinations that include pressure, deadweight, seismic and hydrodynamic loadings. Table 2 provides a tabulation of the Eq.(9) stresses and the allowables used.

The results demonstrate that all of the piping stresses are within allowable Appendix F limits.

A detailed Appendix F analysis of the valves associated with the eleven piping penetrations has not been performed. However, a qualitative assessment of each penetration was performed in order to assess the susceptibility of the subject valves to an overpressurization failure. Based on a review of the isolation valves associated with the eleven penetrations, nine contain inboard and outboard isolation valves of a different type and design. It has been concluded that because of the distinct design characteristics of these valves, the likelihood of simultaneous failure of the inboard and outboard isolation valves is negligible.

The other two penetrations contain isolation valves that are of the same design. Based on inherent manufacturing differences, it is considered highly unlikely that both valves would fail simultaneously, however, this cannot be stated conclusively. Therefore, a

quantitative analysis is being pursued with the valve vendor in order to determine if the subject valves are capable of accommodating the predicted post-LOCA pressures. The results of the analysis will be provided to the NRC by November 30, 2001.

Table 1. Pressure Limits In Accordance With F-1430(a)

Penetration	D_{pipe}	P_{max} (psi)	T_{max} (F)	$P_{allowable}$ (psi)	Comments
X-85A	3”	3010	118	5143	$P_{max} < P_{allowable}$
X-85B	3”	2810	120	5143	$P_{max} < P_{allowable}$
X-86A	3”	4570	139	5143	$P_{max} < P_{allowable}$
X-86B	3”	4400	142	5143	$P_{max} < P_{allowable}$
X-23	4”	2280	133	4360	$P_{max} < P_{allowable}$
X-24	4”	2420	135	4360	$P_{max} < P_{allowable}$
X-17	6”	4600	160	5473	$P_{max} < P_{allowable}$
X-54	8”	2570	114	3059	$P_{max} < P_{allowable}$
X-53	8”	2970	117	3059	$P_{max} < P_{allowable}$
X-56	8”	2570	114	3059	$P_{max} < P_{allowable}$
X-55	8”	3030	114	3059	$P_{max} < P_{allowable}$

Table 2. Stress Limits In Accordance With F-1430(b)

Penetration	D_{pipe}	Hoop Stress (psi)	Mech Stress (psi)	Total Stress (psi)	3.0 S_m or 2.0 S_Y (psi)
X-85A	3”	12193	16399	28592	60000
X-85B	3”	11383	16038	27421	60000
X-86A	3”	18513	10324	28837	60000
X-86B	3”	17824	9122	26946	60000
X-23	4”	10823	20451	31274	60000
X-24	4”	11487	10166	21653	60000
X-17	6”	17636	21853	39489	50200
X-54	8”	17210	13668	30878	60000
X-53	8”	19888	13576	33464	60000
X-56	8”	17210	10255	27465	60000
X-55	8”	20290	10118	30408	60000

Attachment 1 to PLA-5352

50.59 Safety Evaluation Number NL-92-019
Primary Containment Control

SUSQUEHANNA STEAM ELECTRIC STATION
50.59 SAFETY EVALUATION

NUMBER: NL-92-019	
AFFECTED UNIT(S): 1 & 2	
TITLE: Primary Containment Control - EO-000-103	
DCP #:	
PROCEDURE NUMBER: EO-000-103	
OTHER:	

Revision	Preparer	Independent Reviewer
2	[signature]	[signature]
Responsible Supervisor/Date	**PORC Meeting #**	**General Manager SSES/Date**
[signature] 10-8-98	98-10-16	[signature] 10-29-98

Revision	Preparer	Independent Reviewer
Responsible Supervisor/Date	**PORC Meeting #**	**General Manager SSES/Date**

Revision	Preparer	Independent Reviewer
Responsible Supervisor/Date	**PORC Meeting #**	**General Manager SSES/Date**

A copy of the General Manager - SSES APPROVED Safety Evaluation must be forwarded to the Supervisor - Nuclear Licensing, SRC Executive Assistant and the Original to Nuclear Records by the preparer. A copy shall be sent to the Manager System Engineering when required by Form NDAP-QA-0726-02.

SAFETY EVALUATION # NL-92-019

Title: Primary Containment Control - EO-000-103

READ "50.59 SAFETY EVALUATION INSTRUCTIONS" ATTACHMENT C BEFORE ANSWERING EACH QUESTION

I. System/Procedure/Experiment Identification. (Name and Number)

This safety evaluation is performed for EO-000-103, Primary Containment Control.

II. Provide a Description and the Implications of the Proposed Action on the Safe Operation of SSES.

A. Fully describe the action and its purpose.

Description of the Procedure

EO-000-103, Primary Containment Control, is being revised as part of the Severe Accident Management project. With the exception of the items identified in the Technical Adequacy Review Section, Differences from BWROG EPG/SAG, EO-000-103 conform with guidance provided by the BWROG Emergency Procedure and Severe Accident Guidelines, Rev. 1.

EO-000-103, Primary Containment Control provides instructions to maintain primary containment integrity and protect equipment in the primary containment. This is accomplished by control of the following five parameters that impact primary containment integrity: suppression pool temperature, suppression pool water level, primary containment pressure, primary containment combustible gas concentrations and drywell temperature. Each of these parameters is treated independently in a separate leg of the flow chart. Each of these legs will be discussed separately following a descriptions of the entry conditions.

The entry conditions to this procedure are:

Suppression pool temperature greater than 90°F,
Suppression pool water level less than 22',
Suppression pool water level greater than 24',
Drywell pressure greater than 1.72 PSIG,
Primary containment hydrogen concentration greater than 4%, or
Drywell temperature greater than 150°F.

The suppression pool entry conditions are the limits defined in Technical Specification (TS) 3.6.2.1 (CTS & ITS). The drywell pressure entry condition is the isolation and ECCS initiation set point defined in TS 3.3.2 & 3.3.3 (CTS) & 3.3.5.1 (ITS). The primary containment hydrogen concentration entry condition is the alarm setpoint. The drywell temperature entry condition is the highest temperature allowed as an initial condition in the containment analysis (Source Document 6). Therefore these entry conditions are designed to keep the containment parameters within the values assumed in the safety analysis. The procedure response to each entry condition is now discussed.

Suppression Pool Temperature

The suppression pool temperature leg is designed to first maintain the suppression pool temperature within the limits defined by the TS. If the suppression pool temperature cannot be maintained within TS limits then the temperature is maintained below the Heat Capacity Temperature Limit (HCTL). The purpose of the HCTL is to ensure that an unanticipated RPV depressurization will not result in the containment pressure exceeding the Primary Containment Pressure Limit (PCPL) of 65 psig. This limit is based upon the ability to open the primary containment vent valves rather than the containment's ultimate strength which is estimated at 140 psig. Therefore emergency depressurization is directed if the suppression pool temperature cannot be maintained below the HCTL. Events for which the suppression pool cannot be maintained below the HCTL are beyond the design basis of the plant as described in Appendix I of the Design Assessment Report (DAR) (Source Document 1). PP&L has taken two deviations from the generic EPG concerning the HCTL. The first deviation excludes the suppression pool design temperature from the definition of the HCTL and is described in Source Document 2. The second deviation restricts RPV depressurization when the suppression pool temperature cannot be maintained below the HCTL and the reactor power exceeds 5%. This deviation is discussed in Source Document 3.

Suppression Pool Level Control

EO-000-103 is entered when the suppression pool water level cannot be maintained within the TS limits of 22 to 24 feet. If the water level is low the operator is instructed to pump water from the CST to the suppression pool using either HPCI, RCIC or the suppression pool cleanup system. If the suppression pool is high then the operator is instructed to let down the suppression pool to liquid rad waste or the main condenser using either suppression pool filter system or the RHR system. If necessary, containment isolations are bypassed using ES-159/259-002 provided the reactor coolant activity is within TS values. ES-159/259-002 provides instructions to bypass high drywell pressure, reactor low level 3 and low level 2 for Unit 1 and Unit 2, respectively. In addition to installing bypasses in the control structure relay rooms, switches need to be positioned in the control room and valves/breakers need to be positioned in the reactor building. Prior to allowing letdown, the containment is sampled. Letdown is only allowed if the activity meets the criteria established for venting through SGTS. If the actions specified above do not correct the high or low water level, further actions are specified. The particular actions depend upon the particular condition.

Suppression Pool Water Level - Low

A persistent low suppression pool water level can only be the result of a breach of the suppression pool boundary that cannot be isolated. If the breach can be isolated then the step identified above should be able to restore the suppression pool water level to the TS values. Therefore, any actions in this section of the suppression pool level leg are beyond the design basis.

This low level section of the suppression pool level control procedure is designed to prevent or mitigate three particular problems: a reduction in pool heat capacity, a potential degradation or failure of the low pressure ECCS pumping systems from the effects of vortex formation, and a loss of pressure suppression when the HPCI steam exhaust and the LOCA downcomer vents become uncovered. In responding to these potential problems the operator is instructed to reduce or terminate flow from affected pumping systems or rapidly depressurize the RPV prior to significant reduction in heat capacity or a loss of vapor suppression.

Based on BWROG EPG/SAG Rev.1, the Heat Capacity Level Limit (HCLL) was deleted and Operator action to isolate HPCI at a suppression pool level limit of 17 ft was added to EO-000-103. The Heat Capacity Temperature Limit (HCTL) was revised to incorporate the HCLL.

Suppression Pool Water Level - High

This high level section of the suppression pool level leg is designed to prevent the horizontal section of the HPCI and RCIC turbine exhaust lines from flooding. The procedure directs the operator to limit the suppression pool water level to be less than 26 ft. If the suppression pool level cannot be maintained below 26 ft, the horizontal section of HPCI exhaust line may be flooded. The RCIC exhaust line is bounded by this elevation since it's elevation slightly higher than HPCI. The operator is to ensure that HPCI and RCIC are operating and remain operating or start HPCI and RCIC manually. (Source Document 7). The exhaust steam from operation will keep the exhaust line open thus avoiding water hammer.

Finally this section of the suppression pool level leg directs the operator to go to Damage Support Procedure EP-DS-002, RPV and Primary Containment Flooding, if pool level cannot be maintained below 38 ft..

Primary Containment Pressure Control

The purpose of the primary containment pressure control leg is to maintain the containment pressure below the Primary Containment Pressure Limit (PCPL). The PCPL for Susquehanna is the highest containment pressure at which the vent valves can be opened. Before reaching this pressure efforts are directed at maintaining the containment pressure within lower limits.

The first actions are directed at reducing or maintaining the containment pressure below the drywell pressure entry condition of 1.72 PSIG. These actions include maximizing drywell cooling and venting through the SGTS. Venting is only permitted if: (1) the SPING is OPERABLE or alternate sampling is in progress; (2) no SPING alarm condition exists. Therefore venting as allowed by this step is within the design basis. If these actions are ineffective, the operator is directed to initiate wetwell sprays. Suppression chamber spray operation is prohibited, however, if the suppression chamber pressure drops below 0 PSIG, or the pumps needed for spraying the suppression pool are also needed for adequate core cooling. Suppression chamber spray operation is terminated before reaching 0 PSIG to preclude in-leakage of oxygen and avoid exceeding the containment negative design pressure.

If operation of the suppression chamber sprays is unable to prevent the wetwell pressure from exceeding 13 PSIG, then operation of the drywell sprays is authorized. Prior to initiation of the drywell sprays the drywell coolers and recirc pumps must be shut down. Additionally, flow necessary for adequate core cooling cannot be diverted from RPV injection to supply drywell sprays. After the above requirements are fulfilled drywell sprays can be initiated at a flow rate of 1000 to 2800 gpm for at least 30 seconds. This initial throttling provides sufficient vapor to the drywell atmosphere to prevent a rapid drywell depressurization from full drywell spray flow. Subsequent to the throttling period drywell spray flow is increased to the maximum.

Once drywell sprays have been initiated the operator monitors containment pressure. Spray flow is terminated before the drywell pressure drops to zero psig. OP-1/249-004 authorizes the operator to throttle drywell sprays once the maximum flow has been achieved to allow the operator to control the drywell depressurization rate. However drywell sprays are terminated prior to the drywell pressure falling to zero psig.

If the drywell spray operation is ineffective at controlling containment pressure, the RPV is depressurized before the suppression chamber exceeds the Pressure Suppression Limit (PSL). The purpose of the PSL is to ensure that the pressure suppression function is maintained while the RPV is at pressure. This limit addresses three potential containment threats: loss of vapor suppression as a result of the LOCA vents becoming uncovered, loss of the ability to vent the containment as the result of a RPV depressurization exceeding the vent valve opening pressure PCPL, and loss of primary containment integrity as the result of the suppression pool design boundary load being exceeded. PP&L has taken deviation to prevent RPV depressurization on this

limit when the reactor power exceeds 5%. This deviation is discussed in Source Document 4. Finally, when the primary containment cannot be maintained below the PCPL of 65 psig, and the TSC determines that venting is appropriate per EP-DS-004, the primary containment is vented prior to a containment pressure of 65 psig.

Combustible Gas Control

Upon entering the primary containment hydrogen control leg the operator is directed to monitor hydrogen and oxygen concentrations in the drywell and suppression chamber to detect the development of combustible gas concentrations. If either a detectable amount of hydrogen (≥1%) exists, or oxygen concentration is ≥5%, or hydrogen and oxygen concentrations cannot be determined to be below the combustible limits of 6% hydrogen and 5% oxygen, then control is transferred to the Damage Support Procedure, EP-DS-001, Containment Combustible Gas Control.

High Drywell Temperature

EO-000-103 can also be entered as the result of drywell temperature in excess of 150 °F. This leg of the primary containment control procedure is designed to limit the drywell temperature to a value below the 150 °F specified as an initial condition in the safety analysis. The operator is directed to maximize drywell cooling using ES-134/234-001. If a containment isolation has occurred, ES-134/234-001 authorizes bypassing the containment isolation provided the isolation was not caused by a DBA. A containment rad monitor reading of less than or equal to 5 R/hr is used to verify that a DBA was not the cause of the isolation. After maximizing drywell cooling the operator determines if the high drywell temperature invalidates the RPV water level instrumentation by determining if the drywell instrument run is above RPV saturation temperature. If the drywell temperature continues to increase after maximizing drywell cooling, the operator is directed to initiate drywell sprays. The drywell sprays must be terminated before the drywell pressure drops below 0 PSIG. If the drywell sprays are unable to reverse the drywell temperature trend the operator is directed to maintain the drywell temperature below 340 F by ensuring a reactor scram has occurred and initiating a rapid depressurization. The reactor scram is accomplished by a transfer to the RPV Control procedure. After rapid depressurization the operator once again evaluates the RPV water level instrumentation.

Summary of Differences and Deviations

There are three substantive differences between PP&L's Primary Containment Control procedure and the BWROG EPG/SAG for primary containment control. These differences exist because: the particular EPG/SAG step could not be implemented using the equipment installed at Susquehanna, the Susquehanna procedure was enhanced beyond what is required by the BWROG EPG/SAG Rev. 1, or a disagreement exists between PP&L and the BWROG concerning appropriateness of certain procedural actions. These differences are presented in the following Table along with the rational for the difference.

Item Number	EPG/SAG Caution or Step Number	Description	Difference Between BWROG Guidance and EO-000-103
1	SP/T-3	Depressurize RPV when suppression pool temperature cannot be maintained below HCTL.	RPV depressurization when the suppression pool temperature cannot be maintained below the HCTL is being restricted if the reactor power exceeds 5% power. PP&L calculations indicate that severe core damage may occur if the reactor is operated at low pressure when the reactor power exceeds 5%.
2	SP/L2.1	Depressurize RPV when suppression pool water level cannot be maintained above downcomer openings.	RPV depressurization when the suppression pool level cannot be maintained above 12' is being restricted when the reactor power is greater than 5%. See SP/T-3 for rational.
3	PC/P-3	Depressurize when suppression chamber pressure cannot be maintained below pressure suppression limit.	RPV depressurization when the containment pressure cannot be maintained below the PSL is being restricted when the reactor power exceeds 5%. See SP/T-3 for the rational.

B. Identify all the components that will be affected.

All systems that can be used to remove heat from the suppression pool, the drywell, and the suppression chamber airspace and associated support equipment. All systems that can be used to control the primary containment pressure and the associated support equipment. All equipment that can be used to add water to or remove water from the suppression pool and the associated support equipment. All equipment that can be used to detect hydrogen or oxygen and to determine combustible gas concentration in the primary containment and the associated support equipment.

C. List Safety Functions of affected components.

EO-000-103 deals with controlling suppression pool temperature and level, drywell temperature, primary containment pressure, and primary containment hydrogen and oxygen. Therefore, the components affected by EO-000-103 are involved with controlling these functions.

D. Describe potential effects on Safety Functions.

As described in Section II.A, there are no adverse effects on safety functions of the affected components.

E. Describe the affect on 80-10 Systems.

The proposed procedure changes, which are within the design basis, do not create a new release path from the primary containment to the environment. The procedure steps, which bypass containment isolation, are outside of the design basis.

III Does the proposed action **increase the probability of occurrence or the consequences** of an accident or malfunction of equipment important to safety, as previously evaluated in the SAR? (Include specific reference to SAR sections that are applicable.)

YES ☐	NO ☒

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

EO-000-103, Primary Containment Control does not increase the probability of occurrence or the consequence of an accident or equipment malfunction important to safety as previously evaluated in the FSAR. The actions specified in this procedure are taken after the incidence of the initiating event. Therefore they cannot increase the probability of an accident or malfunction previously analyzed in the FSAR. In fact this procedure implements all of the operator actions in the FSAR or the DAR (Source Document 1) which are assumed to occur when demonstrating compliance of the containment design with the regulations. A review of FSAR Sections 6.2.1, Primary Containment Functional Design, 6.2.2, Containment Heat Removal Systems, 6.2.5, Combustible Gas Control, Chapter 15 Accident Analysis and Appendix I of the DAR (Source Document 1) Suppression Pool Response to SRV Discharge was carried out to identify what operator actions were assumed in the accident analyses. The following Table identifies the actions identified in the FSAR review and the corresponding EO-000-103 operator actions.

Comparison of Operator Actions Assumed in the Safety Analysis with EO-000-103 Guidance.

Action	Operator Action Assumed in FSAR	Corresponding EO-000-103 Guidance
1	Initiation of Drywell sprays 600 seconds or more after a recirculation system piping rupture. This action is not required to meet the design basis.	Drywell sprays are authorized when the wetwell pressure exceeds 13 psig.
2	Initiation of suppression pool cooling 600 sec after a recirculation system piping system rupture	Suppression pool cooling is authorized if the suppression pool temperature cannot be maintained below 90 °F.
3	Initiation of suppression pool sprays before the containment pressure reaches 30 PSIG	Wetwell sprays are authorized if the drywell pressure cannot be maintained below 1.72 psig.
4	Monitoring combustible gas concentrations in the primary containment	Hydrogen concentration is an EOP entry condition. Additionally, steps in the hydrogen control leg require monitoring hydrogen and oxygen concentrations.
5	Initiation of the recombiners when the combustible gas levels exceed 3.%,	Initiation of recombiners is authorized before the hydrogen concentration reaches 2%.
6	Initiation of containment purging through SGTS post LOCA.	Authorization is provided to use purge through the SGTS to maintain drywell pressure less than 1.72 psig.

These actions are all called out in the procedures and are within the design basis. Additionally actions are called out to restore the containment parameters to the values assumed in the safety analysis. Some of these steps require that isolations and interlocks be bypassed to accomplish the step. These steps are listed below along with the ES procedure used to effect the bypass, the prerequisites that are required to implement the ES procedure, and the rationale supporting why implementation is consistent with the design basis.

STEP	ES PROCEDURE	IMPLEMENTATION PREREQUISITES	DESIGN BASIS JUSTIFICATION
SP/L-10	ES-1/259-002	Reactor coolant activity within TS limits	Pool letdown is only allowed if coolant activity is within TS limits. Pool activity is inferred from reactor coolant activity.
DW/T-2	ES-1/234-001	Drywell temperature cannot be maintained below 150°F and containment rad level ≤ 5 R/hr	Drywell cooling is a closed system that does not communicate with the drywell atmosphere. Drywell cooling is designed to isolate during a DBA. No credit for drywell cooling is taken in the DBA analysis. The ES is implemented after the initiating event subsequent to the electrical load sheds and sequencing associated with DBA are complete. The ES is only implemented if the Rad level ≤ 5 R/hr verifying that no DBA source term exist.

The actions specified in the ES procedures do not invalidate any of the assumptions in the FSAR or the DAR (Source Document 1). All of these actions are designed to restore the containment parameters to the TS limits or initial conditions of the safety analyses. Therefore they do not increase the consequence of an accident.

IV Does the proposed action create a **possibility for an accident or malfunction of a different type** than any evaluated previously in the SAR? (Include reference to specific SAR sections applicable.)

YES ☐	NO ☒

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

The actions proposed in EO-000-103, Primary Containment Control, do not create the possibility for an accident or malfunction of a different type than evaluated previously in the FSAR. As described above the actions directed by this procedure for conditions that are in the design basis are designed to either: implement the actions specified in the safety analyses described in Chapters 6 and 15 of the FSAR or Appendix I of the DAR (Source Document 1) or restore the containment process parameters to normal values. Actions, such as primary containment venting, which are inconsistent with the design bases analysis are only implemented when the accident has progressed beyond the design bases. These actions include:

- Venting the primary containment when the pressure cannot be maintained below 65psig.
- Terminating HPCI irrespective of adequate core cooling when the suppression pool water level cannot be maintained above 17'.

A discussion of why these actions are beyond the design bases follows.

Venting the primary containment when the pressure cannot be maintained below 65 psig

The primary containment is designed to condense steam that is discharged from the RPV and to contain radioactivity following a postulated accident. The achievement of these two functions requires that the primary containment pressure boundary be maintained. Therefore, willful loss of primary containment

integrity is inconsistent with the design basis of the containment. When reviewing the accident analysis in the FSAR and the DAR (Source Document 1) one finds that the containment pressure never exceeds the design value of 53 PSIG for the initiating events and single failures postulated. Therefore these particular procedural steps would only be executed if the accident progressed beyond those events as analyzed in the FSAR and the DAR (Source Document 1). The Susquehanna IPE (Source Document 5) was reviewed to determine the nature of events in which the containment pressure would exceed the design value. In all cases multiple failures were required in addition to the initiating event. Therefore no events have been identified in which the containment pressure exceeds 65 psig and still have the plant operating within its design basis. Since venting is only considered once the plant has proceeded beyond the design basis, it cannot create an accident or malfunction different than what is analyzed in the FSAR.

Terminating HPCI irrespective of adequate core cooling when the suppression pool water level cannot be maintained above 17'

Step SP/L-6 directs the operator to isolate HPCI irrespective of adequate core cooling if the suppression pool water level cannot be maintained above 17'. If the HPCI exhaust sparger becomes uncovered, the HPCI exhaust steam will directly pressurize the wetwell air space and possibly fail the containment on overpressure. Therefore the procedures require that HPCI be isolated. A suppression pool water level that cannot be maintained above 17' can only occur if the suppression pool pressure boundary is breached and unable to be isolated. This event however is outside the design basis. Therefore, implementation of step SP/L-6 does not create the possibility, within the plant design basis, of an accident or malfunction of a different type than any evaluated in the FSAR.

V Does the proposed action reduce the **margin of safety** as defined in the basis for any Technical Specification? (Include reference to specific Technical Specification sections that are applicable.)

YES ☐	NO ☒

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

The operator actions proposed in EO-000-103 do not involve changing any Technical Specification basis. Therefore, the margin of safety as defined in the basis of any Technical Specification is not reduced.

VI Does the proposed action involve a change in a Technical Specification?

YES ☐	NO ☒

If "YES", NDAP-QA-0731 "Technical Specification Changes" applies. A "YES" answer does not preclude activity up to a point just before it would physically affect the functioning of the plant.

Provide a discussion of the basis and criteria used in arriving at the above conclusion. If appropriate, describe the extent of activity and why it should be allowed to proceed prior to the Technical Specification change.

Implementation of EO-000-103 does not involve a change to any Current or Improved Technical Specification.

VII Does the proposed action create the need to make an application for amendment to the license other than to Appendix A?

YES ☐	NO ☒

Provide a discussion of the basis and criteria used in arriving at the above conclusion.

No amendment to the licensing is being made through this procedure.

SOURCE DOCUMENTS:

1. Susquehanna Steam Electric Station, Design Assessment Report.

2. Safety Evaluation Eliminating the Suppression Pool Design Temperature from the Heat Capacity Limit. (NL - 92-018)

3. Safety Evaluation Prohibiting RPV Depressurization When the Suppression Pool Temperature Cannot Be Maintained Below the Heat Capacity Temperature for Events Where the Reactor Power Is Greater Than 5%. NL - 92 - 016)

4. Safety Evaluation Prohibiting RPV Depressurization When the Suppression Chamber Pressure Cannot Be Maintained Below the Pressure Suppression Limit When the Reactor Power Exceeds 5%. (NL - 92 - 017)

5. Susquehanna Steam Electric Station, Individual Plant Evaluation, NPE-91-001, December 1991.

6. Effect of Initial Drywell Temperature on Peak Drywell Pressure and Temperature During Limiting LOCAs for Susquehanna Units 1 & 2. S. K. Rhow, GE Nuclear Energy. GE-NE-B13-01805-89 Class III October 1996.

7. Safety Evaluation 93-3070/71. HPCI Pump Suction Auto Transfer to Suppression Pool Logic Elimination (Units 1 & 2)

Technical Adequacy Review for E0-000-103

This Technical Adequacy Review addresses actions specified by EO-000-103 that are outside the plant design basis. This review includes application of the placement criteria for steps that have been moved to the TSC. Additionally, justification for steps that deviate from the BWROG EPG/SAG Rev. 1 is provided. Next, compliance to the procedural defense in depth criteria is demonstrated. Finally, a response to the questions raised by NRC are addressed. Detailed discussions for these items are presented below.

Actions Outside Plant Design Basis

Nearly all the actions specified in E0-000-103 are implemented while conditions remain within the SSES design basis. However, there is a case where this procedure authorizes the use of plant equipment for suppression pool level control when the normal methods of pool level control are not available or inadequate. In other cases, steps in E0-000-103 deal with certain primary containment parameter which exceeds its design limit. The procedure steps involving actions outside of design basis are discussed below.

SP/L-4 The normal suppression pool level control methods are to use the suppression pool cleanup system. When the normal methods are not available or inadequate for maintaining pool level above 17 ft, EO-000-103 authorizes the use of HPCI and RCIC to add water to the suppression pool. If HPCI and RCIC are required for adequate core cooling, then the use of HPCI and RCIC for suppression pool level control is not allowed. Allowing HPCI and RCIC to be used provides additional methods to meet the intent of the procedure step, while a higher priority is placed on assuring adequate core cooling than maintaining suppression pool level.

SP/L-5 Before the suppression pool level drops to 17 ft, the operator is instructed to go to RPV control procedure. This step is adequate, because entering EO-000-102 assures that the reactor can be scrammed and shutdown prior to isolating HPCI and rapidly depressurizing the RPV. These actions may be required as the suppression pool level continues decreasing.

SP/L-6 If the suppression pool level cannot be maintained above 17 ft, the HPCI system is to be isolated irrespective of adequate core cooling. This action is to address the concern of a uncovered HPCI turbine exhaust sparger. If suppression pool level were to drop below 17 feet with HPCI operating, the turbine exhaust would discharge directly into the suppression chamber airspace causing subsequent overpressurization of the primary containment. Isolation of HPCI is compensated for by instructions in RPV Control. Entry into RPV Control is directed before level drops to 17 feet. RPV Control gives level control guidelines and prioritizes injection systems.

SP/L-7, SP/L-8 & SP/L-9 When the suppression pool level cannot be maintained above 12 ft, and reactor power is less than 5%, the reactor is to be rapidly depressurized. As the downcomers will uncover at pool level of 12 ft, any steam discharged from the RPV into the drywell may bypass the suppression pool and pressurize the primary containment to unacceptable levels. Rapid depressurization is restricted to reactor power less than 5%, because severe core damage may occur with the reactor operated at low pressure and power exceeding 5%. However, if the suppression pool level is already below 12 ft, downcomer vents are actually becoming uncovered and the pressure suppression function no longer exists. Therefore, rapid depressurization is required irrespective of power level. Depressurizing RPV through SRVs is safe, because T-quenchers do not uncover until pool level is below 5 ft.

SP/L-10, and SP/L-11 The HPCI turbine exhaust line begins to flood at a pool level of 25'7". To maintain the suppression pool level below 26 ft will prevent HPCI exhaust line from flooding. Since removal of water from the suppression pool may be prevented by isolation signals, permission is given in ES-159/259-002, Suppression Pool Letdown / Containment Venting Isolation Bypass, to bypass isolation signals to: (1) Suppression Pool Filter Pump, and (2) RHR Shutdown Cooling Mode.

If the suppression pool level cannot be maintained below 26 feet and HPCI system is not running, the horizontal section of the turbine exhaust line might be flooded up to isolation check valve F049. When this occurs, the length of water column in the exhaust line increases by about 25 feet. Due to inertial effects, a high turbine exhaust pressure will develop as this column of water is expelled upon auto restart of the turbine. This may cause the HPCI pressure

relief-diaphragms to rupture and render the system inoperable. Therefore, if HPCI is not running, the Operator is directed to start HPCI. If HPCI is running, it is allowed to continue to operate.

SP/L-12 As discussed in the Placement Criteria Section, a suppression pool water level of 38' can only occur given a deliberate effort to increase the containment water level as proceduralized in EP-DS-002. Thus this high level section of the suppression pool level control is transferred to the Damage Support Procedure, EP-DS-002, RPV and PC Flooding when the pool level cannot be maintained below 38 ft.

DW/T-6 When the drywell temperature cannot be maintained below 340 °F, EO-000-102 is entered at step RC-1. This assures that the reactor will be scrammed and shutdown before RPV depressurization is started. Rapidly depressurizing the RPV minimizes further energy release from the reactor to the drywell. In turn, this action reduces any continued drywell temperature increase.

Placement Criteria Evaluation

As certain primary containment parameters reach their preset limits, instructions for operation are to be transferred from EO-000-103 to the Damage Support Procedures at the following three steps:

SP/L-12 When suppression pool level cannot be maintained below 38 ft, go to EP-DS-002, RPV and PC Flooding.

PC/P-11 Before primary containment pressure reaches 65 psig, vent primary containment in accordance with EP-DS-004, Primary Containment Venting.

PC/G-6 When H_2 concentration reaches 1%, or O_2 concentration equal to or greater than 5%, or H_2/O_2 concentrations cannot be determined below combustible limits, go to EP-DS-001, Containment Combustible Gas Control.

Deterministic Criterion Evaluation

Since the actions in Steps SP/L-12 and PC/P 11 are not assumed in the FSAR Design Basis Events, the deterministic criterion is not applicable.

In EO-000-103 the concentrations of H_2 and O_2 in the drywell and suppression chamber are monitored using H_2/O_2 analyzer or by primary containment sampling for hydrogen and oxygen. In EP-DS-001, Containment Combustible Gas Control, this action is also performed. In addition, when H_2/O_2 cannot be determined to be below the combustible limits, all recombiners are to be shutdown. Because both H_2/O_2 monitoring and using H_2/O_2 recombiners are the actions assumed in the FSAR, transferring control from EO-000-103 to EP-DS-001 would not violate the actions assumed in the Design Basis Events.

Mechanistic Criterion Evaluation

The mechanistic criterion evaluation is to determine whether or not the action is required prior to activation of the Technical Support Center (TSC). The evaluation for each of the aforesaid steps is presented below.

Step SP/L-12

The normal water source for adding water into the suppression pool is the condensate storage tank (CST). The CST water level is normally maintained at 225,000 gallons. If this amount of water is added into the suppression pool, the pool level will rise from 24 feet to 29.5 feet. Therefore, the pool level cannot increase from 24 feet to 38 feet in any credible accident event. The suppression pool level may approach 38 ft, only if the damage support procedure, EP-DS-002, RPV and PC Flooding, has been executed from EO-000-102, EO-000-113, or EO-000-114. Therefore, when the suppression pool level approaches 38 ft, the TSC should already be functional.

Step PC/P-11

The venting procedure is expected to take about 40 minutes to complete. This is because the procedure includes preparatory actions to: 1) align for nitrogen purge; 2) align for alternate RPV injection; 3) block open a high energy room door; 4) cross-tie instrument air, service air and containment instrument gas; 5) align reactor building HVAC dampers; and 6) bypass isolation signals to the vent valve. Initiation of the venting procedure is expected to occur at an elevated containment pressure of approximately 60 psig, but not to exceed 65 psig, at which point the pressure differential across the vent valves may prevent them from opening . If an accident scenario causes containment pressure to rise from 60 psig to 65 psig in less than 40 minutes and the operator does not contact TSC for venting recommendation until pressure reaches 60 psig, then venting will not be possible. ATWS, large break LOCA without vapor suppression, and high pressure vessel failure scenarios will likely cause containment pressure to rise from 60 psig to 65 psig in less than 40 minutes. Small and medium break LOCAs without vessel failure or with low pressure vessel failure will likely cause containment pressure to take more than 40 minutes to rise from 60 psig to 65 psig .[1]

Most probable events and times to reach 60 psig are given in the following table:

	Time to Reach 60 psig in Containment (Hrs.) Transient	LOCA
No Vessel Failure	16.1	15.0
Low Pressure Vessel Failure	13.8	11.5

The two 'no vessel failure' cases are loss of decay heat removal events. The loss of decay heat removal causes containment pressure to increase. The RPV will be depressurized at a cooldown rate of approximately 100 °F/hr. Reactor coolant temperature will reach its minimum value equal to the saturation temperature at containment pressure in less than four hours. At this time, the suppression pool temperature will be well above 120 °F. Since the containment will be at an elevated pressure the minimum reactor coolant temperature will be much higher than 200 °F. A suppression pool temperature of 120°F with RPV coolant higher than 200°F constitutes a Site Area Emergency per criteria #16 of the Emergency Plan. The TSC will be staffed within one hour of activation. The time span from the TSC being fully staffed to containment pressure reaching 60psig is more than 10 hours. Therefore, the TSC should be functional long before the containment pressure reaches 60 psig.

The first 'low pressure vessel failure' case is a short term SBO transient with failure of high pressure injection. The core is uncovered in about 33 minutes. Coreplate dryout occurs in less than 12 minutes later. The second 'low pressure vessel failure' case is a recirculation line break followed by initiation of ADS at -129' with no injection into the RPV. In this case core uncovery and coreplate dryout take place in less than a minute. An RPV water level below top-of-active fuel for greater than 3 minutes constitutes a Site Area Emergency per criteria # 4 of the emergency Plan. The TSC will be staffed within one hour of activation. The time span from the TSC being fully staffed to containment pressure reaching 60 psig is more than 12 hours for the first 'low pressure vessel failure' case and more than 10 hours for the second case. Therefore, the TSC should be functional long before the containment pressure reaches 60 psig.

Step PC/G-6

If the hydrogen and oxygen concentrations in the primary containment reach sufficiently high level, combustion may occur. Before concentrations of both gases reach the flammable level, procedure control should be transferred from EO-000-103 to EP-DS-001, Containment Combustible Gas Control. The combustible limit of oxygen concentration is 5%. In an inerted containment oxygen can appear from either the radiolysis of water or the interaction of core debris and concrete

[1] Safety Evaluation NL 98-036 EP-DS-004 "Primary Containment and RPV Venting

The radiolysis of water takes place in the reactor vessel. If the pressure boundary of the primary loop is breached, oxygen and hydrogen produced from water radiolysis can be released into the primary containment from the reactor vessel. Because the water radiolysis is a very slow process, during a DBA LOCA event, it takes more than one day for oxygen and hydrogen concentrations to reach the level requiring H_2/O_2 recombiner operation as shown in Fig. 6.2-50 of the FSAR. Shortly after the initiation of a DBA LOCA event the TSC is activated. The TSC will be fully functional before oxygen concentration reaches the combustible limit.

Core concrete interaction takes place on drywell floor after core debris has relocated to the drywell floor and sufficient water is not on the floor to quench the core debris. Based on the results of several severe accident calculations with vessel lower head rupture the shortest time between core uncovery and lower head rupture is 124 minutes following a Large Break LOCA with failure of all injection. The TSC should be activated shortly after core uncovery and fully functional prior to the onset of core concrete interaction.

Risk Significance Criterion Evaluation

The risk significance criterion evaluation is to determine the effect of relocating the action outside of control room on probability of success and plant damage frequency. At Susquehanna the actions in the emergency procedures of the damage support program are to be performed by operators in the control room. However, the TSC takes over the responsibility of decision making from the on-shift organization. Therefore, transfer from EOPs to EPs of the damage support program does not affect either the probability of success or the plant damage frequency.

Differences Between EO-000-103 and BWROG Guidelines

This Section provides the justification for differences between the BWROG EPG/SAG and EO-000-103. The following Table lists those steps where differences exist. A justification for each of these differences is provided subsequent to this Table.

Item Number	EPG/SAG Caution or Step Number	Description	Difference Between BWROG Guidance and EO-000-103
1	Caution-1	Caution stating that RPV LVL INSTR may be used only when temp. near INSTR RUN below RPV SAT temp.	An override in EO-000-103.
2	SP/L-2.2	Maintain SUPP POOL LVL above HPCI exhaust line.	Add the use of HPCI and RCIC in EO-000-103.
3	SP/L-2.1	SUPP POOL LVL cannot be maintained above downcomer openings, RAPID DEPRESS is . REQ'D	Add restriction of PWR ≤ 5% in EO-000-103.
4	PC/P-2	Use drywell spray initiation limit in BWROG Appendix C	Use drywell spray initiation limit developed by PPL.
5	PC/P-3	When suppression chamber pressure cannot be maintained below PSP RAPID DEPRESS is REQ'D	Add restriction of PWR ≤ 5% in EO-000-103
6	SP/T-3	When SUPP POOL TEMP and RPV PRESS cannot be maintained below HCTL Rapid DEPRESS is required.	Add restriction of PWR ≤ 5%.
7	SP/L-3.1	Maintain SUPP POOL LVL below SRV Tail Pipe LVL Limit	Not included in EO-000-103
8	PC/P-4	Vent irrespective of offsite radioactivity release rate	Venting in accordance with EP-DS-004.
9	SP/L-3.2	Maintain Suppression Pool Level below drywell to wetwell vacuum breakers	Vacuum breakers are above the level at which EO-000-103 transfers to EP-DS-002.

Item 1 - Caution 1 and DW/T-3 concerning the use of RPV level instrument run temperature

The complexity of BWROG EPG/SAG Caution 1 and the importance associated with the validity of RPV water level readings require that this concern be expressed in an override instead of a Caution. Since instrument run temperature in the drywell above 212°F will not be reached before average drywell temperature reaches 150°F, this override will always be evaluated when the BWR EPG would require its use. It is translated into an action in Step DW/T-3. Parts 2 and 3 of BWR EPG Caution #1 which define Minimum Indicated Levels and Maximum Run Temperatures are addressed by SSES-EPG Caution 1 and step DW/T-3. Reactor Building instrument run temperature concerns addressed by BWROG EPG/SAG Caution 1 are presented in the SC/T section of the Secondary Containment Control Guideline.

Item 2 - Maintain suppression pool water level above the top of HPCI exhaust line

A list of methods which can be used to add water to the suppression pool is added in Step SP/L-4. This step adds the use of HPCI and RCIC on minimum flow except as required to assure adequate core cooling. Authorization to use HPCI and RCIC is given here since the Suppression Pool Cleanup System alone has been unsuccessful at maintaining pool level when reading this step. Allowing HPCI and RCIC to be used provides two additional methods to meet the intent of this step.

Item 3 - If suppression pool level cannot be maintained above the downcomer openings rapid RPV depressurization is required

The instruction, "...but only if reactor power is less than 5%." is added to Step SP/L-8. RPV depressurization due to a projected suppression pool water level below 12 feet is only performed if reactor power is less than 5% because of concerns for fuel damage caused by instabilities at low pressure operation and power levels above 5%. Item 6 provides additional explanation.

Item 4 - When suppression chamber pressure exceeds 13 psig drywell spray is required

According to EPG/SAG, drywell sprays can be initiated only if the drywell temperature and pressure being within the Drywell Spray Initiation Limit (DWSIL). This DWSIL was developed by using the methodology provided in Appendix C of EPG/SAG. This DWSIL prohibits use of drywell sprays when they are required, because the recipe is overly simplistic. The DWSIL developed by PP&L allows drywell spray initiation for any credible drywell temperature and pressure condition when drywell sprays are required. The only restriction of the PPL DWSIL is to limit the spray flow to between 1000 and 2800 gpm for the first 30 seconds. This restriction is incorporated in Step PC/P-7.

Item 5 - When suppression chamber pressure cannot be maintained below the Pressure Suppression Pressure, rapid depressurization is required

The restriction, "...but only if reactor power is less than 5%" is added in Step PC/P-9 of EO-000-103. Because emergency depressurization with the power greater than 5% may result in core damage from prompt critical reactivity insertions and containment pressurization is expected during severe ATWS events, depressurization based on the PSL is restricted if the reactor power is greater than 5%. (Source Document 4)

Refer to Item 6 for further explanation and justification of this difference.

Item 6 - When suppression pool temperature and RPV pressure cannot be maintained below HCTL rapid depressurization is required

The restriction, "...but only if reactor power is less than 5%" is added in Step SP/T-3 of EO-000-103. This restriction is consistent with the key assumption made in deriving the HCTL that the reactor is shutdown. Depressurizing the RPV when the reactor is not shutdown violates this key assumption made in deriving HCTL. Additionally as discussed in Safety Evaluations NL 92-016 and -017 operation on a critical reactor at low pressure may cause core damage. Therefore depressurization of on the HCTL is not included in EO-000-103.

Item 7 - Maintain suppression pool water level below the SRV Tail Pipe Level Limit

The SRV Tailpipe Level Limit is not limiting until suppression pool levels above 38'. Since parameter control transfers to RPV and Primary Containment Flooding at 38', this step is not applicable. The RPV is depressurized prior to entering EP-DS-002

Compliance to the Procedural Defense In Depth Criteria

Criterion P1 - No procedure shall have adverse consequences in the case of additional equipment failures beyond those occurring initially.

Additional equipment failures are categorized by the following safety functions: suppression pool cooling, suppression pool level control, containment pressure control, hydrogen control, and drywell temperature control.

Additional Equipment Failures that Impact Suppression Pool Cooling

Some equipment in the suppression pool cooling loop may fail. Then the operator can use the other RHR loop for suppression pool cooling. However, this option is restricted, if RHR pumps are needed for adequate core cooling. This restriction assures that maintaining adequate core cooling takes precedence over suppression pool cooling.

The operators may not be able to maintain suppression pool temperature and RPV pressure below HCTL because equipment failure in the suppression pool cooling loop. Then EO-000-103 calls for RPV rapid depressurization. This action is restricted to reactor power less than 5% to assure adequate reactivity control.

Additional Equipment Failures that Impact Suppression Pool Level Control

The normal suppression pool water level control method is to use the suppression pool cleanup system. If this system is inoperable, the HPCI and RCIC systems may be used in the minimum flow mode to add water to the suppression pool from the CST. However, if HPCI and RCIC are required for adequate core cooling, they can not be used for suppression pool level control. The use of HPCI is also restricted when suppression pool level cannot be maintained above 17 ft. The purpose of this restriction is to protect the suppression chamber from over pressurization due to HPCI exhaust sparger uncovering. Therefore, this procedure step does not have adverse consequence on adequate core cooling or containment integrity.

When suppression pool level cannot be maintained above 12 ft and reactor power is less than 5%, the operator is directed to rapidly depressurize the reactor before the downcomer openings are uncovered. The restricting condition of reactor power being less than 5% assures no adverse consequence on reactivity control.

Additional Equipment Failures that Impact Containment Pressure Control

The drywell coolers, the SGTS, and the containment spray system are the equipment to be used for containment pressure control. Failures of the containment pressure control equipment can result in high containment pressure. When suppression chamber pressure cannot be maintained within PSL limit and reactor power is less than 5%, the RPV is to be rapidly depressurized. The restricting condition of reactor power being less than 5% assures no adverse consequence on reactivity control.

If the primary containment pressure reaches 65 psig, the operator is authorized to vent the primary containment in accordance with EP-DS-004 to maintain pressure below 65 psig. EP-DS-004 recommends venting when release rates are within TS limits; recommends venting with caution when offsite dose projections are less than 10CFR100; recommends not venting when doses are greater than 10CFR100. This assures that the primary containment integrity is maintained by using the vent valves before primary containment pressure exceeds the design opening pressure of the vent valves without an unnecessary release of radioactivity.

Additional Equipment Failures that Impact Hydrogen Control

The H_2/O_2 analyzers are used to monitor hydrogen and oxygen concentration in the primary containment. If they are not available, then Chemistry is notified to determine the concentration of these gases by sampling and analysis. This action does not interact with any plant system and can not have any adverse consequence.

Additional Equipment Failures that Impact Drywell Temperature Control

If, due to drywell cooling equipment failure, the RPV level instrument run temperature exceeds the RPV Saturation Temperature, then the RPV level instrument cannot be used. This step identifies the usability of the RPV level instrument and enhances RPV level control.

If, due to drywell cooling equipment failure, the drywell temperature cannot be maintained below drywell design temperature (340 °F), EO-000-103 instructs the operator to go to EO-000-102 to orderly shutdown the reactor then rapidly depressurize it. This assures no adverse consequence on reactivity control.

Criterion P2 - The necessary anticipatory actions shall be performed to avoid loss of additional equipment but shall not degrade the existing situation

EO-000-103 is implemented to control various primary containment parameters. Therefore, this procedure is reviewed to determine what actions are taken to protect equipment used for controlling these parameters. Many prerequisites and precautions are reviewed and discussed in the operating procedures for each system. These items are not reviewed in this TAR. Specific actions which apply to severe accidents are reviewed as follows.

Step SP/T-6 The instruction directs the operator to rapidly depressurize the RPV, if suppression pool temperature and RPV pressure cannot be maintained below HCTL. This action prevents the primary containment pressure from exceeding 65 psig to protect the operability of containment vent valves.

Steps SP/L-11 When suppression pool level cannot be maintained below 26 ft, if the HPCI system is not running, then it is to be started manually to prevent auto start with a flooded exhaust line. This action protects the operability of HPCI system.

Criterion P3 - The necessary anticipatory actions shall be performed to permit successful response to potential additional failures, but shall not degrade the existing situation

EO-000-103 is implemented to control various primary containment parameters. Therefore, this procedure is reviewed to determine what actions are taken to permit successful response to additional failure of equipment used for controlling various primary containment parameters. Each of these actions is discussed below.

Step SP/T-3 If suppression pool temperature is rising due to loss of suppression pool cooling, EO-000-102 will be entered before pool temperature reaches 110 F. Entering EO-000-102 at Step RC-1 assures that, if possible, the reactor is scrammed and shutdown by control rod insertion. Entry into RPV control also meets the prerequisite for RPV rapid depressurization, which may be required to avoid exceeding the Heat Capacity Temperature Limit.

Step PC/P-7 If suppression chamber spray could not be started, drywell spray is started when suppression chamber pressure exceeds 13 psig. This spray operation reduces drywell pressure and temperature. When the drywell pressure drops below the suppression chamber pressure by 0.5 psi, the vacuum breakers will begin to open. This will allow gas and vapor flow into the drywell from the suppression chamber to reduce suppression chamber pressure.

Step PC/G-6 When hydrogen and oxygen concentrations in the drywell or suppression chamber cannot be determined by any means, it must be assumed that concentration levels in excess of those required to support combustion are present. Therefore, EO-000-103 directs the operator to contact the TSC to enter EP-DS-001, Containment Combustible Gas Control to take appropriate actions, such as start containment sprays and venting, to effectively control the combustible gases.

Response to NRC Questions

The BWROG sent the EPG/SAG to the NRC for their review. As part of this review, the NRC generated a number of questions. These questions were forwarded[2] to the individual owners and are being addressed accordingly. There are two questions which apply to primary containment control. These two questions and their resolutions are discussed below.

1. Emergency Depressurization After PSL Exceeded, (NRC Question 17)

The primary effects of RPV rapid depressurization on suppression pool are temperature increase and level increase due to mass and energy addition from the reactor vessel and the suppression pool boundary load resulting from SRV actuation. As the suppression pool temperature rises, more vapor can be added into the suppression chamber airspace through the evaporation process at pool surface. In addition, more energy can be added into the airspace from the suppression pool through the heat transfer process at pool surface. The mass and energy addition through the evaporation process and the heat transfer process will increase the suppression chamber airspace pressure, Psc. However, the mass and energy addition rate through these two processes are very slow. As a result, the Psc increase due to rapid RPV depressurization is very slow. The results of a PP&L calculation for small break LOCA and steam bypass event has verified that the effect of RPV depressurization on Psc is secondary.[3]

The primary effect of RPV rapid depressurization on containment structural integrity is the suppression pool boundary load, Psrv, during SRV actuation. This hydrodynamic load is generated by steam discharge from the T-quenchers, which are located at a level of 3.5 ft in the suppression pool. This load is one of the limiting factors for the Pressure Suppression Pressure Limit (PSL) according to Appendix C of the BWROG EPG/SAG. However, the structural failure

[2] Letter from D.B. Matthews, USNRC to K.P. Donovan, BWROG Emergency Procedures and Severe Accident Guidelines, April 2, 1997.
[3] PP&L Calculation No. EC-thyd-1025.

point of the primary containment is estimated to be at elevation of 26.3 ft in the suppression chamber[4]. This point is above the suppression pool surface. Therefore, this hydrodynamic load does not need to be included in the PSL. Another limiting factor for PSL is the Primary Containment Pressure Limit of 65 psig. The primary containment ultimate pressure capability is estimated to be 140 to 160 psig. Hence there is a safety factor of 2 to 2.5. Based on the above discussion, it can be concluded that the PSL is very conservative.

When the suppression chamber condition approaches PSL, the most effective method to stop suppression chamber pressurization is to rapidly depressurize the RPV with ADS actuation. Within 15 to 20 minutes from ADS actuation, the RPV will be at about the ambient pressure and is no longer a pressure source for containment pressurization[5]. There will be only a slight increase in Psc within this time span, because the effect of RPV blowdown on Psc is secondary as explained above. After the RPV is completely depressurized, the containment pressure will stop increasing and may begin to decrease if energy and/or mass are being removed from the primary containment by certain mitigating measures such as drywell spray initiation. Since RPV rapid depressurization after exceeding PSL does not have a deleterious impact, it is not being restricted once PSL is exceeded.

2. Use of Containment Sprays (NRC Question 12)

The NRC has voiced concerns with the drywell spray initiation limit curve in their review of the BWROG EPG Rev. 4 changes and Severe Accident Management Guidelines. Specifically the NRC states:

The Drywell Spray Initiation Limit (DWSIL) is calculated assuming a 32 °F spray water, no humidity in the drywell atmosphere, and no flow through the vacuum breakers. This would produce a curve that does not reflect typical conditions, and that may be so restrictive that the drywell sprays cannot be initiated when needed, e. g., prior to vessel breach.

PP&L has similar concerns with the assumptions used to develop the DWSIL curve as does the NRC. These concerns address artificial limitations that seem to prevent spraying the drywell when needed. Because of in-house concerns and a desire to addresses those of the NRC, calculations were performed to determine an appropriate DWSIL curve. These calculations are documented in PP&L recorded calculation EC-THYD-10216.

The CONTAIN code was used to develop the DWSIL for Susquehanna. CONTAIN is a state of the art containment analysis code developed by the USNRC. The NRC has used CONTAIN in a number of their containment building evaluations, which includes Susquehanna. CONTAIN was used in conjunction with the following assumptions to develop an appropriate DWSIL.

1. At the time of drywell spray initiation, the suppression pool water is at 40°F. The RHR heat exchanger cold leg inlet temperature is constant and equal to 32°F.

2. The wetwell and drywell are at the same pressure when drywell sprays are initiated.

3. The wetwell-to-drywell vacuum breakers are operable,

4. The initial mass of water vapor in the drywell is 563.3 lbm, which is equal to the amount of water vapor mass at the initial condition (120°F, 48% relative humidity) that is imposed for DBA LOCA analysis.

5. The initial wetwell airspace temperature equals 90°F with 100% relative humidity.

6. The amount of noncondensable gases in the primary containment is determined by the initial pressure and temperature in the drywell and wetwell. All the noncondensable gases are represented in one case by nitrogen and in an other case by hydrogen. These two gases bound the possible range of effects.

[4] PP&L Memo from J.A. Swankoski to C. Kukielka, "Susquehanna Primary Containment Ultimate Pressure Capacity", August 7, 1992.
[5] EC-RISK-0514, Impact of Power Uprate on the 1991 Susquehanna IPE.
[6] EC-THYD-1021, Drywell Spray Initiation Limit, Rev.3.

The results of these calculations were used to generate the PPL DWSIL. This
DWSIL provides protection against containment damage from exceeding the design differential pressure, yet does not unnecessarily restrict operation of the drywell sprays. Furthermore, the DWSIL developed by PP&L addresses the concerns raised by the NRC.

Based on the PPL DWSIL, drywell sprays can be initiated without concern at any drywell condition with credible temperature and pressure combinations, if the spray flow is within the range of 1000 gpm to 2800 gpm. After 30 seconds, the drywell atmosphere contains sufficient vapor to allow full drywell spray flow. Based upon these considerations the DWSIL can be formulated in the following statement:

Initiate drywell sprays between 1000 gpm and 2800 gpm for the first 30 seconds. After 30 seconds no restrictions apply to drywell spray flow.

3. Terminating Drywell Spray before the drywell pressure drops to 0 psig. (NRC Question 10)

The NRC has expressed concerns that terminating the drywell sprays before the containment pressure drops to 0 psig will result in inleakage of oxygen, thus de-inerting the containment. This concern is inpart based upon the reactor building to containment vacuum breakers in the Mark I containment design and the potential for the containment pressure to drop below 0 psig due to either instrument drift or operator error. This is not at issue at Susquehanna which is a Mark II containment which does not have reactor building to containment vacuum breakers.

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SAFETY EVALUATION SUMMARY

Title: Primary Containment Control - EO-000-103

No. NL-92-019

Description of Change: EO-000-103, Primary Containment Control is being revised as part of the Accident Management Implementation Project. These procedure steps in EO-000-103 are consistant with the guidance in the BWROG EPG/SAG Rev. 1. Differences between EO-000-103 and the BWROG EPG/SAG are provided in the Technical Adequacy Review attached to this Safety Evaluation.

This procedure provides the operator with guidance for controlling the following containment parameters: suppression pool temperature, suppression pool level, drywell temperature, containment pressure and containment combustible gas concentration. The actions specified to control these parameters are consistent with the analysis in the FSAR for events within the design basis and include: initiation of suppression pool cooling, initiation of containment sprays, initiation of the hydrogen recombiners, purging the primary containment through SGTS. Authorization to bypass isolation and prevent the HPCI suction swap from the pool is provided as needed to control events. However this authorization is only provided for events outside the plant design basis.

Major revisions to the procedure include: transferring much of the combustible gas control procedure to EP-DS-001, replacing the drywell spray initiation limit with instructions to throttle drywell spray flow, transfer of much of the high suppression pool level leg to the EP-DS-002, and transferring the decision process for Primary Containment venting to EP-DS-004. Justifications for these changes are provided in the Technical Adequacy Review.

SUMMARY

A. Actions specified in EO-000-103 occur after an initiating event. Therefore these actions cannot increase the frequency or the probability of an event analyzed in the FSAR. EO-000-103 implements the operator actions specified in the FSAR analysis. Actions which bypass isolation signals or allow operation of HPCI with suction from the CST when the pool level exceeds 24' are only authorized for events that are outside the plant licensing basis.

B. The actions proposed in EO-000-103, Primary Containment Control, do not create the possibility for an accident or malfunction of a different type than evaluated previously in the FSAR. As described above the actions directed by this procedure for conditions that are in the design basis are designed to either: implement the actions specified in the safety analyses described in Chapters 6 and 15 of the FSAR or Appendix I of the DAR (Source Document 1) or restore the containment process parameters to normal values. Actions, such as primary containment venting, which are inconsistent with the design bases analysis are only implemented when the accident has progressed beyond the design bases.

C. The operator actions proposed in EO-000-103 do not involve changing any Technical Specification basis. Therefore, the margin of safety as defined in the basis of any Technical Specification is not reduced.

Attachment 2 to PLA-5352

A Methodology for Calculating Penetration Piping Heatup and Pressurization during a Postulated Design Basis Accident

A METHODOLOGY FOR CALCULATING PENETRATION PIPING HEATUP AND PRESSURIZATION DURING A POSTULATED DESIGN BASIS ACCIDENT

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August 16, 2001

PURPOSE

The purpose of this paper is to present the methodology and results of an analysis used to predict the heat up and pressurization of piping with entrapped fluid during a postulated loss of coolant accident. This analysis is performed to respond to concerns in Generic Letter GL 96-06 about the potential for thermally-induced over-pressurization of containment penetration piping.

SUMMARY

Generic Letter GL 96-06 raises the concern that during a postulated Loss of Coolant Accident (LOCA) piping inside containment will be heated beyond its normal service condition and water trapped between closed isolation valves will expand due to the heating and pressurize the line, challenging the pipe integrity. A methodology for analyzing the heatup and pressurization of fluid entrapped in a closed piping system is presented. The model accounts for axial and radial conduction through the pipe wall, convection heat transfer to the fluid inside and to the ambient region outside, radiation heat transfer to/from surrounding structures, condensation on the exterior of the pipe, and the effects of pipe insulation. Piping embedded within the containment wall is assumed to be perfectly insulated (no heat loss to the concrete) and can not expand due to pipe heat up or pressurization. Piping inside of containment is exposed to drywell ambient conditions and a steam environment due to reactor blowdown from the postulated LOCA. No credit is taken for drywell sprays to cool the containment atmosphere or penetration piping. Condensation heat transfer to the piping inside the drywell is accounted for when a steam environment exists; natural convection heat transfer is considered otherwise. Piping outside of containment is exposed to secondary containment ambient conditions and heat transfer between the piping and secondary containment is accounted for. Heat is transferred from the pipe to the entrapped fluid by natural convection and the time-dependent fluid heat up is calculated. As its temperature rises the fluid expands and pressurizes the pipe. The heat up and pressurization of the pipe causes the pipe to expand providing a degree of pressure relief. The actual pipe pressure is calculated taking into account the balance between the fluid and pipe expansion. Validation against experimental data demonstrates that the methodology predicts conservatively high values for pipe pressure.

Fluid heatup and pipe pressurization resulting from a postulated small break accident (SBA) is analyzed. The SBA is the containment temperature design basis accident for Susquehanna as discussed in FSAR Chapter 6. The design basis SBA analysis provides a bounding drywell temperature profile by imposing a 6-hour reactor blowdown at maximum drywell superheat conditions. The resulting drywell temperature of 340F is assumed to be sustained for the entire 6-hour blowdown period. For long-term decay heat removal beyond 6 hours, the current analysis employs a single RHR heat exchanger with the maximum design basis heat sink temperature. MSIV's are assumed closed for the entire duration of the event and RHR is the only source of decay heat removal.

Realistically, the operators would initiate suppression pool cooling within 10 to 20 minutes following the LOCA. Emergency procedures authorize the operation of the drywell sprays once the drywell pressure exceeds 13 psig, which occurs shortly after the SBA. The operators will initiate a controlled cool down once the RPV parameters are stabilized. It is expected that the reactor will drop to the shutdown cooling interlock pressure within 3 to 4 hours. Therefore the SBA temperature profile bounds what would realistically occur under accident conditions.

Eleven containment penetrations are analyzed for thermally-induced pressurization. Comparison of Susquehanna Unit 1 and Unit 2 showed that for a majority of penetrations the piping geometry of the two units is essentially identical and a typical piping layout could be analyzed. In only one case, penetration X-23, was the piping within the drywell different between units. For this penetration, the piping run for both units was analyzed. A table is included presenting the maximum pressure and temperature predicted for each penetration along with the time at which they occur. Also presented is the piping pressure at 24 hours into the event, to provide a perspective on the timing of the heatup.

METHODOLOGY

A generic methodology is developed to model the thermal response of a fluid-filled pipe exposed to various ambient regions. The model accounts for axial and radial conduction through the pipe wall, convection heat transfer to the fluid inside and to the ambient region outside, radiation heat transfer to/from surrounding structures, condensation on the exterior of the pipe, and insulation on the pipe. Time dependent ambient conditions can be set in up to 100 ambient zones. Zone 1 is reserved for the drywell air space and is the only ambient zone where condensation is considered. In the case of a postulated LOCA scenario, the drywell is assumed to contain a steam environment and condensation heat transfer is taken into account. The methodology is implemented in the FORTRAN program PTRAP (Pipe Thermal Response Analysis Program) developed and maintained on PPL computers.

ASSUMPTIONS

The following major assumptions are employed to facilitate a simplified methodology which provides a conservative estimate of the fluid heatup:

1. The pipe thermal response is two-dimensional (radial & axial) and is uniform in the azimuthal direction.
2. A lumped capacitance model for fluid heatup is used. The duration of the transients is sufficiently long that natural convection currents will maintain fluid mixing.
3. To maximize the radiation heatup effect, structures surrounding the pipe are considered blackbodies and no credit is taken for radiation absorption or scattering by the steam environment.
4. No credit is taken for contact resistance between the insulation and pipe in the heat transfer model.
5. The thermal capacitance of the insulation is neglected, since its mass is significantly less than the pipe mass.
6. No heat loss occurs from the pipe to the concrete containment wall.
7. Only the thinnest schedule pipe expands when pressurized [1]. As an additional conservatism, thermal expansion is also neglected on the thicker schedule piping.
8. Pipe embedded within the containment wall does not expand.
9. No credit is taken for enhanced compressibility of the entrapped fluid due to the presence of non-condensable gas which may be dispersed within the process fluid.
10. The condensation model assumes a pure steam environment to maximize the condensation heat transfer to the pipe.

Case-specific inputs impose further assumptions which simplify and add conservatism to the analysis.

Model Overview & Features

A pipe layout to be analyzed is broken into segments, each segment having the same properties of:

- Orientation (horizontal/vertical)
- pipe diameter and material
- insulation type & thickness
- ambient zone

- radiation source

Individual segments are then broken down into elements for the purpose of numerical discretization. An energy balance is performed on each pipe element taking into account convection to the internal fluid, convection and radiation to the exterior region, and conduction with the two adjacent pipe elements. A sample pipe layout for a containment penetration is shown in Figure 1.

The piping model is extended beyond the penetration pipe to account for the heat conduction from pipes upstream and downstream of the isolation valves. Piping beyond the isolation valves is assumed empty to minimize the heat sink. Figure 2 presents a schematic of a pipe segment discretized into a number of elements. The entire piping system is composed of N elements. The governing equations are derived for a typical element j shown in Figure 2.

The inside surface area of the elements is exposed to the entrapped fluid. The outside surface area is either covered with insulation or is directly exposed to ambient and radiation conditions. Elements at the boundary of the piping system are designated boundary elements (j=1, j=N) and are not exposed to either the fluid or ambient conditions. The boundary elements provide conduction boundary conditions for the elements j=2 and j=N-1. For the GL 96-06 evaluation, insulated boundaries are considered to minimize heat loss from the piping.

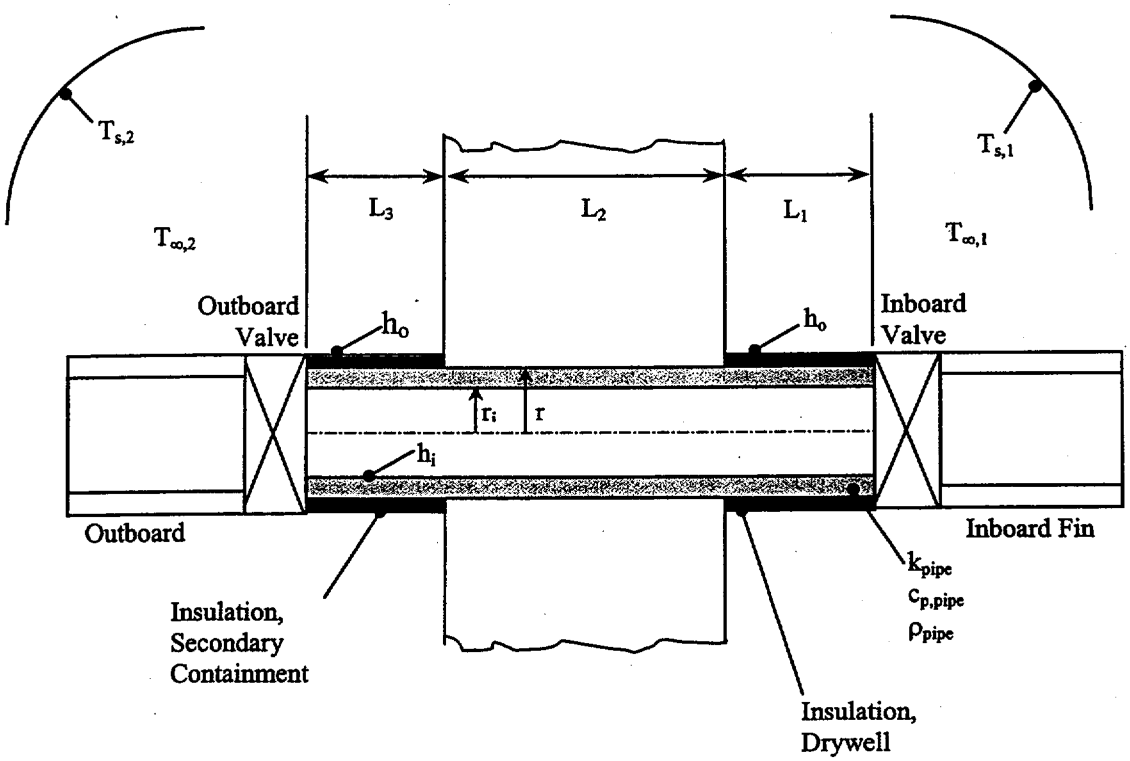

Figure 1: Sample Pipe Layout

Following is a list of model features available in the methodology:

- Time-dependent drywell spray in Ambient Zone 1,
- Condensation heat transfer in Ambient Zone 1,
- Condensation correlations for vertical and horizontal pipes,
- Radiation heat transfer between the outside of the pipe and surrounding blackbody structures,
- Convection heat transfer with the ambient air space outside the pipe,
- Convection heat transfer with the entrapped fluid within the pipe,
- Conduction heat transfer in the axial and radial direction,
- Insulated or uninsulated pipe,
- Lumped capacitance fluid heatup,
- Time-dependent ambient temperatures,
- Time-dependent drywell pressure to specify steam conditions,
- Axial variation in pipe and insulation size, materials, and heat transfer properties,

A second-order Runga-Kutta method [2] was used to integrate the solution in time. Runga-Kutta is a well-accepted method for integration using an explicit algorithm for marching in time. The time step used must be taken small enough to assure the solution is time-step insensitive to a level appropriate for the accuracy required.

Governing Equations – Pipe Heat Transfer Model

Each pipe element (*j*) represents a control volume for which an energy balance is performed. Energy enters and leaves the control volume by conduction, convection, and radiation heat transfer. Energy is stored by the thermal capacitance of the pipe material. The energy balance for a typical element given by Eq. (1) is a balance between the net conduction heat transfer with neighboring elements, convection heat transfer with the internal fluid and exterior environment, radiation heat transfer with the surrounding structures, and stored energy in the pipe volume, reflected in the pipe element temperature response.

$$\dot{E}_{stored} = \dot{E}_{conduction} + \dot{E}_{convection} + \dot{E}_{radiation} \quad . \tag{1}$$

Positive values on the right hand side of Eq. (1) represent energy transfer into the volume. The components of the energy balance for an uninsulated pipe element *j* are:

$$\dot{E}_{stored} = m_j c_p \left. \frac{\partial T}{\partial t} \right|_j , \tag{2}$$

$$\dot{E}_{conduction} = k_p^+ A^+ \left.\frac{\partial T}{\partial x}\right|_{j+1,j} - k_p^- A^- \left.\frac{\partial T}{\partial x}\right|_{j,j-1}, \tag{3}$$

$$\dot{E}_{convection} = U_{o,j} A_{o,j} (T_{o,j} - T_j) + U_{i,j} A_{i,j} (T_f - T_j), \tag{4}$$

$$\dot{E}_{radiation} = \varepsilon_{o,j} \sigma A_{o,j} (T_{s,j}^4 - T_j^4). \tag{5}$$

where

$$U_{o,j} = \frac{1}{\dfrac{1}{h_{o,j}} + \dfrac{\ln(r_{o,j}/\bar{r}_j) r_{o,j}}{k_{p,j}}}, \tag{6}$$

and

$$U_{i,j} = \frac{1}{\dfrac{1}{h_{i,j}} + \dfrac{\ln(\bar{r}_j/r_{i,j}) r_{i,j}}{k_{p,j}}}, \tag{7}$$

with the following variables defined as:

m_j = mass of pipe element j,
c_p = specific heat of pipe material,
t = time,
$k_{p,j}$ = thermal conductivity of pipe material for element j,
$h_{i,j}$ = convection heat transfer coefficient on pipe inner surface,
$h_{o,j}$ = convection heat transfer coefficient on pipe outer surface,
r_i = inner-radius of the pipe,
r_o = outer-radius of the pipe,
$\bar{r}_j$ = mid-radius of the pipe,
A_j = element j pipe cross-sectional area = $\pi(r_o^2 - r_i^2)_j$,
$A_{i,j}$ = element j pipe inner surface area = $(2\pi r_i L)_j$,
$A_{o,j}$ = element j pipe outer surface area = $(2\pi r_o L)_j$,
T_j = temperature of pipe element j,
$T_{o,j}$ = ambient temperature for element j,
$T_{s,j}$ = surrounding temperature for element j radiation,
$\varepsilon_{o,j}$ = emissivity of outer surface of element j,
σ = Stefan-Boltzmann constant (0.1714×10^{-8} Btu/hr-ft^2-R^4),

$$k_p^+ = \frac{1}{2}(k_{p,j} + k_{p,j+1}),$$

$$k_p^- = \frac{1}{2}(k_{p,j} + k_{p,j-1}),$$

$$A^{+} = \frac{1}{2}(A_j + A_{j+1}),$$

$$A^{-} = \frac{1}{2}(A_j + A_{j-1}).$$

The spatial derivatives in Eq. (3) are approximated as:

$$\left.\frac{\partial T}{\partial x}\right|_{j+1,j} = \frac{(T_{j+1} - T_j)}{\Delta x^{+}}, \tag{8}$$

$$\left.\frac{\partial T}{\partial x}\right|_{j,j-1} = \frac{(T_j - T_{j-1})}{\Delta x^{-}}, \tag{9}$$

where Δx^{+} = distance between axial centerpoint of elements j and $j+1$,
Δx^{-} = distance between axial centerpoint of elements j and $j-1$.

For an insulated pipe, the treatment is slightly different because it is the insulation surface temperature that transfers heat with the surroundings. In the case of an insulated pipe, the insulation surface temperature $T_{ins,j}$ is found by considering the conduction heat transfer across the insulation thickness from the pipe, and balancing it with the convection and radiation at the insulation surface, bearing in mind that the thermal capacitance of the insulation mass is neglected. For an insulated pipe, the following equation is used to account for heat transfer with the surroundings for pipe element j:

$$\dot{E}_{convection} = U_{o,j}A_{o,j}(T_{ins,j} - T_j) + U_{i,j}A_{i,j}(T_f - T_j). \tag{10}$$

For insulated pipes, the outer surface area is defined by the insulation radius $r_{ins,j}$ as:
$A_{o,j} = 2\pi r_{ins,j} L_j$.

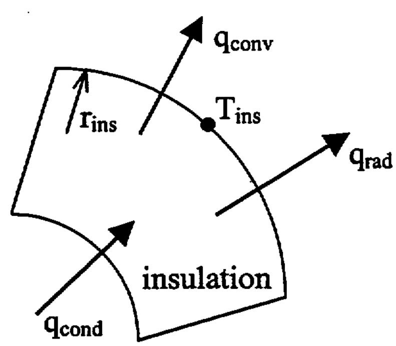

Figure 3: Insulation Energy Balance

For the energy balance at the surface of the insulation, we will neglect the energy storage term in the insulation, as its mass is typically much smaller than the pipe mass. The energy balance on the insulation is thus:

$$\dot{E}_{conduction} + \dot{E}_{convection} + \dot{E}_{radiation} = 0\,, \tag{11}$$

where:

$$\dot{E}_{conduction} = U_{o,j} A_{o,j} (T_{ins,j} - T_j)\,, \tag{12}$$

$$\dot{E}_{convection} = h_{o,j} A_{o,j} (T_{ins,j} - T_{o,j})\,, \tag{13}$$

$$\dot{E}_{radiation} = \varepsilon_{o,j} \sigma A_{ins,j} (T_{ins,j}^4 - T_{s,j}^4) \cong h_{r,j} A_{ins,j} (T_{ins,j} - T_{s,j})\,, \tag{14}$$

$$U_{o,j} = \frac{1}{\dfrac{\ln(r_{ins,j} / r_{o,j}) r_{ins,j}}{k_{ins,j}} + \dfrac{\ln(r_{o,j} / \bar{r}_j) r_{ins,j}}{k_{p,j}}}\,. \tag{15}$$

In Eq. (14), the radiation heat transfer is linearized using a so-called radiation heat transfer coefficient $h_{r,j}$ as outlined in Ref. [3], where:

$$h_{r,j} = \varepsilon_{o,j} \sigma (T_{ins,j} + T_{s,j})(T_{ins,j}^2 + T_{s,j}^2)\,. \tag{16}$$

Substituting Eqs. (12) - (16) into Eq. (11) yields an expression for the insulation surface temperature:

$$T_{ins,j} = \frac{h_{r,j} T_{s,j} + U_{o,j} T_j + h_{o,j} T_{o,j}}{h_{r,j} + U_{o,j} + h_{o,j}}\,. \tag{17}$$

With the insulation surface temperature from Eq. (17), the energy balance (1) for insulated pipe element j can be solved using Eq. (10) for the convection component along with Eqs. (2), (3) and (5).

Governing Equations – Fluid Heat Transfer Model

The fluid exchanges heat with the pipe through convection across an internal thermal boundary layer. The fluid is treated as a lumped capacitance and mixing allows all pipe elements to see the same fluid temperature, while the fluid sees elements of differing

temperature. The energy balance on the fluid mass yields the equation for the fluid temperature and is expressed as:

$$m_f c_{p,f} \frac{dT_f}{dt} = 2\pi \int_0^L h_i(x) r_i(x)(T_{pipe}(x) - T_f)dx. \tag{18}$$

Equation (18) is evaluated numerically by discretizing over interior elements *j=2, N-1* as:

$$m_f c_{p,f} \frac{dT_f}{dt} = 2\pi \sum_{j=2}^{N-1} h_{i,j} r_{i,j} (T_j - T_f) \Delta L_j , \tag{19}$$

where ΔL_j = the length of element *j*. Equation (19) is coupled to the energy balance for each pipe element *j* and is solved along with the pipe temperatures in the Runga-Kutta solution.

Heat Transfer Coefficients

The natural convection heat transfer coefficients $h_{o,j}$ and $h_{i,j}$ in the previous section are calculated at each time step from accepted heat transfer correlations. The heat transfer coefficients are calculated from the standard relation

$$h = \frac{Nu_L k}{L}, \tag{20}$$

where k is the thermal conductivity of the surrounding fluid evaluated at the film temperature, L is the characteristic length of the respective segment (segment length for a vertical pipe, diameter for a horizontal pipe), and the Nusselt number Nu_L is evaluated from published heat transfer correlations.

For a vertical cylinder the following correlation for Nusselt number is applicable [4]:

$$Nu_L = \left\{ 0.825 + \frac{0.387 Ra_L^{1/6}}{\left[1 + (0.492/\Pr)^{9/16}\right]^{8/27}} \right\}^2 , \tag{21}$$

while for a horizontal cylinder the following correlation for Nusselt number is applied [5]:

$$Nu_D = \left\{ 0.60 + \frac{0.387 Ra_D^{1/6}}{\left[1 + (0.559/\Pr)^{9/16}\right]^{8/27}} \right\}^2 , \tag{22}$$

where the Rayleigh number is defined

$$Ra_x = \frac{g\beta(T_s - T_\infty)x^3}{\nu\alpha}. \tag{23}$$

In the above equations, fluid properties are:

Pr = Prandlt Number,
β = coefficient of thermal expansion,
α = thermal diffusivity,
ν = kinematic viscosity.

The above properties are evaluated at the film temperature $T_f = (T_s + T_l)/2$, where T_s is the surface temperature and T_l is the fluid temperature.

Equations (21) and (22) are used to evaluate the natural convection heat transfer coefficient on the inside of the pipe where the working fluid is water and on the outside of the pipe where the working fluid is air.

A sensitivity study was done to assess the impact of a larger external heat transfer coefficient to assure uncertainty in the calculated values would not significantly change the predicted peak pipe pressure. The results show that one to two orders of magnitude increase in the natural convection heat transfer coefficient will have relatively little effect (<2% change) on the calculated pressure.

Condensation Model

Condensation heat transfer can be orders of magnitude higher than natural convection heat transfer and it is important to take condensation on cold piping into consideration when modeling postulated LOCA events. The current methodology allows for condensation heat transfer in Ambient Zone 1, which is the drywell. The heat transfer is modeled using a heat transfer coefficient based on film condensation of water vapor on either a horizontal or vertical pipe. The presence of non-condensable gas in a steam environment significantly reduces the condensation heat transfer [6]. Therefore, a pure steam environment is assumed in the condensation model.

On a vertical tube in a saturated steam environment the average condensation heat transfer coefficient based on the length L of the tube is [7]:

$$\bar{h}_L = 0.943\left[\frac{g\rho_l(\rho_l - \rho_v)k_l^3 h'_{fg}}{\mu_l(T_{sat} - T_s)L}\right]^{1/4}, \tag{24}$$

where, g=32.2 ft/s^2, ρ is density (lb$_m$/ft^3), μ is the dynamic viscosity (lb$_m$/ft-s), k is the thermal conductivity (Btu/s-ft-F), and the subscripts l and v indicate liquid and vapor respectively. Liquid properties are evaluated at the film temperature $T_f = (T_{sat} + T_s)/2$, and the modified latent heat of vaporization is [3],[7]:

$$h'_{fg} = h_{fg} + 0.68 c_{p,l}(T_{sat} - T_s) . \tag{25}$$

On a horizontal tube, the average condensation heat transfer coefficient based on the tube diameter D is [8]:

$$\bar{h}_D = 0.729 \left[\frac{g\rho_l(\rho_l - \rho_v)k_l^3 h'_{fg}}{\mu_l(T_{sat} - T_s)D} \right]^{1/4} . \tag{26}$$

Condensation takes place if the surface temperature T_s is less than the vapor saturation temperature T_{sat} at the existing drywell pressure P_{dw}. If ambient drywell temperature falls below $T_{sat}(P_{dw})$, air has been introduced to the drywell via the vacuum breakers. In this case, the code assumes 100% relative humidity and uses the ambient temperature $T_{o,j}$ as the vapor saturation temperature in calculating convection heat transfer to the pipe. The partial pressure of vapor $P_{sat}(T_{o,j})$ is used with $T_{o,j}$ to evaluate vapor/liquid properties in this case.

For cases where the environment surrounding the pipe contains superheated steam, the mass/energy balance used to derive the above expressions is revisited, and Eq. (24) and Eq. (26) are adjusted to consider the superheat values of the vapor density ρ_v, and the actual temperature T_o in place of the saturation temperature T_{sat}. In addition, the total enthalpy change of the fluid $(h_{superheat} - h_f)$ is used in place of h'_{fg}.

Pipe Pressurization Model

As a fluid heats up, its density decreases and its volume expands. In the current problem, the fluid is constrained by the isolated pipe which exerts a force on the fluid as it tries to expand and pressurizes the system. If the pipe is taken as rigid and the fluid as purely incompressible, a volume expansion due to fluid heatup would produce an infinite pressure increase and piping failure due to overpressurization. In reality, the water is slightly compressible which will accommodate a small amount of thermal expansion. The pipe volume will also expand due to (1) thermal expansion with the heatup and (2) elastic/plastic straining of the pipe. If we consider that at each time step the pipe comes to an equilibrium state, then the volume increase due to fluid thermal expansion will be balanced by the compressibility of the fluid plus the pipe expansion due to thermal and pressure effects. A balance of these affects will produce the actual pressure experienced during fluid heatup.

Fluid Expansion

The pressure response of a fluid in a closed (rigid) container undergoing a thermal expansion can be expressed as [9]:

$$dP = \left(\frac{\partial P}{\partial T}\right)_V dT + \left(\frac{\partial P}{\partial V}\right)_T dV, \qquad (27)$$
$$= \frac{\beta}{\kappa} dT - \frac{1}{\kappa}\frac{1}{V} dV,$$

where β = coefficient of cubical expansion of water (1/R)
κ = isothermal compressibility of water (1/psi).

Re-writing Eq. (27) to relate the change in volume to pressure and temperature changes yields

$$\left.\frac{\Delta V}{V}\right|_F = \beta \Delta T_F - \kappa \Delta P, \qquad (28)$$

where Eq. (28) is written in algebraic form for implementation in a numerical integration over finite time steps.

Equation (28) provides a relation for the fluid volume change resulting from thermal expansion due to the fluid temperature differential ΔT_F and compression due to the pressure differential ΔP. Now, if we consider the increase in pipe volume, less pressure will be required to offset the fluid thermal expansion.

Pipe Thermal Expansion

The pipe will expand in both the radial and axial directions as it heats up. For the current analysis, we will assume that the piping embedded within the containment penetration is anchored in place and will not expand. In addition, pipe expansion will only be accounted for in the lowest schedule (thinnest wall) pipe, as this will expand more readily than the thicker-walled pipe [1].

The thermal expansion of the pipe due to a change in temperature ΔT_P is a product of its characteristic dimension, the coefficient of thermal expansion for the pipe material, and the temperature differential ΔT_P. Therefore, for the pipe [10]

$$\Delta r = r_1 \alpha \Delta T_P \quad , \qquad (29)$$

$$\Delta L = L_1 \alpha \Delta T_P \quad , \tag{30}$$

where the subscript "1" refers to the radius r and length L from the previous time step.

The increase in volume due to pipe thermal expansion is calculated as

$$\Delta V_{P,T} = V_2 - V_1 = \pi(r_1 + \Delta r)^2 (L_1 + \Delta L) - \pi r_1^2 L_1 \,. \tag{31}$$

Substituting Eq. (29) and (30) into Eq. (31) and neglecting higher order terms yields the expression for the change in pipe volume due to thermal expansion relative to the original volume $V=\pi r_1^2 L_1$

$$\frac{\Delta V_{P,T}}{V} = 3\alpha \Delta T_P \,. \tag{32}$$

Pipe Expansion due to Pressure Straining

The pipe will also expand in both the radial and axial directions due to the pressure force on the inside surface. This response is a function of the material properties and pipe size, and is given as [11]

$$\Delta r = \frac{\Delta P r_1^2}{Et}\left(1 - \frac{\nu}{2}\right) \quad , \tag{33}$$

$$\Delta L = \frac{\Delta P r_1 L_1}{Et}(0.5 - \nu) \,, \tag{34}$$

where the subscript "1" refers to conditions at the previous time step, $\Delta P = P_2 - P_1$ is the process pressure increase inside the pipe, t is the pipe wall thickness, E is Young's modulus and ν is Poisson's ratio for the pipe material.

The increase in pipe volume is calculated as

$$\Delta V = V_2 - V_1 = \pi(r_1 + \Delta r)^2 (L_1 + \Delta L) - \pi r_1^2 L_1 \,. \tag{35}$$

Substituting Eqs. (33) and (34) into Eq. (35) and neglecting higher order terms yields the expression for the change in pipe volume due to pressure straining relative to the original volume $V=\pi r_1^2 L_1$

$$\frac{\Delta V_{P,P}}{V} = \frac{\Delta P r_I}{Et}\left(\frac{5}{2} - 2\nu\right). \tag{36}$$

The total change in pipe volume due to the heatup ΔT_P and pressurization ΔP is obtained by summing Eqs.(32) and (36) to yield

$$\frac{\Delta V_{P,P}}{V} + \frac{\Delta V_{P,T}}{V} = \frac{\Delta P r_I}{Et}\left(\frac{5}{2} - 2\nu\right) + 3\alpha\Delta T_P. \tag{37}$$

As stated above, only a portion of the pipe will expand; the pipe length embedded within the containment penetration and connected higher schedule piping is assumed not to expand. The expression for pipe volume expansion in Eq. (37) will be multiplied by the fraction of pipe allowed to expand L_e/L_{tot}, where L_e is the pipe length which can expand and L_{tot} is the total pipe length containing fluid. The final expression for the pipe volume expansion is

$$\left.\frac{\Delta V}{V}\right|_P = \frac{L_e}{L_{tot}}\left[\frac{\Delta P r_I}{Et}\left(\frac{5}{2} - 2\nu\right) + 3\alpha\Delta T_P\right]. \tag{38}$$

Equation (38) conservatively neglects the thermal expansion of the thicker schedule piping.

Pipe Internal Pressure Calculation

To calculate the pipe pressure due to the expansion of the entrapped fluid accounting for fluid compressibility, pipe thermal expansion, and pipe pressure strain, we will equate the fluid volume response with the pipe volume response and calculate the pressure for a given temperature which will yield a change in pipe volume equal to the change in fluid volume.

Equation (28) gives the expression for the fluid volumetric change due to fluid heatup of ΔT_F and pressurization ΔP. Equation (38) gives the expression for the change in pipe volume due to pressurization ΔP and pipe heatup ΔT_P. Equating Eqs. (28) and (38) gives

$$\frac{L_e}{L_{tot}}\left[\frac{\Delta P r_I}{Et}\left(\frac{5}{2} - 2\nu\right) + 3\alpha\Delta T_P\right] = \beta\Delta T_F - \kappa\Delta P. \tag{39}$$

The pressure differential ΔP is the change in pressure with time and is defined as

$$\Delta P = P_2 - P_1 \tag{40}$$

where P_1 is the pressure at the previous time step. Solving for the current pressure P_2 in Eq. (40), yields

$$P_2 = P_1 + \frac{\beta\Delta T_F - 3\gamma\alpha\Delta T_P}{\frac{\gamma r_1}{Et}\left(\frac{5}{2} - 2\nu\right) + \kappa}, \tag{41}$$

where

γ	=	L_e/L_{tot},
P_1	=	fluid pressure at last time step (psi),
β	=	coefficient of cubical expansion of water (1/R),
κ	=	isothermal compressibility of water (1/psi),
r_1	=	pipe inner radius at the previous time step (in),
E	=	Young's modulus for the pipe material (psi),
ν	=	Poisson's ratio for the pipe material,
α	=	coefficient of thermal expansion for the pipe material (1/R),
ΔT_F	=	$T_{F,2}-T_{F,1}$ = fluid temperature rise from the last time step (R),
ΔT_P	=	$T_{P,2}-T_{P,1}$ = pipe temperature rise from the last time step (R).

The material properties are evaluated at the average temperature between the current time step and previous time step.

After solving Eq. (41) for P, the value for pipe radius is updated for use in the next time step using Eqs. (29) and (33).

$$r_2 = r_1 + \Delta r = r_1 + \frac{\Delta P r_1^2}{Et}\left(1 - \frac{\nu}{2}\right) + r_1\alpha\Delta T_P. \tag{42}$$

Validation

The PTRAP methodology for calculating pipe heatup and pressurization outlined above is validated by comparing code predictions with experimental data and analytic solutions of sample problems. A suite of problems exercising the various models was run and compared to analytic solutions to assure that the associated physical processes are properly modeled. Experimental data from an Electric Power Research Institute (EPRI) test program on thermally-induced pipe pressurization was used to benchmark the current methodology over the range of process pressure, temperature, and pipe radius applicable to the Susquehanna evaluation. The test facility used by EPRI to investigate thermally-induced pipe pressurization [1] was modeled and the PTRAP predictions were compared to the EPRI test data. Results of the EPRI benchmark cases demonstrate that the PTRAP predictions of pipe pressure bound the EPRI measured data in all cases, verifying that the PTRAP methodology will calculate a conservative value of pipe pressure. The

benchmarks to the EPRI data are summarized below; detailed comparisons are available in Reference [12].

The EPRI test program considered heatup and pressurization of the following pipe configurations: (1) Test #1: a 3 inch schedule 40 stainless steel pipe, (2) Test #2: a combined 3 inch schedule 40/schedule 80 stainless steel pipe, and (3) Test #3: an 8 inch schedule 40 carbon steel pipe. The pipe length is 5 pipe diameters. Data for temperature, pressure, and radial and axial displacement was recorded for each test. The PTRAP models were constructed so that the fluid temperature tracked that in the EPRI tests. This is accomplished by setting the ambient temperature in the PTRAP model equal to the measured fluid temperature in the EPRI tests and setting the convection heat transfer coefficients sufficiently large to provide little thermal resistance between the ambient and fluid regions. These validation cases compared EPRI pipe pressures and pipe radii time histories to PTRAP predicted values for the given fluid temperature response.

A fourth validation case was performed in which an average pipe-wall temperature from EPRI Test #3 is simulated in the PTRAP model and the code calculates the heat transfer coefficient and the fluid temperature inside of the pipe. This case was used to test the methodology for calculating the internal heat transfer coefficient and fluid temperature response.

In all cases, the PTRAP methodology predicted pipe pressures which are higher than the EPRI experimental data. Pressure was over-predicted by greater than 300 psid except for Test #2, when both the schedule 40 and schedule 80 pipe were allowed to expand. In this case, the margin dropped to 100 psid. When only the thinner wall schedule 40 pipe was allowed to expand the predicted pressure increased, verifying that additional conservatism is built into the current assessment by considering only expansion of the thinnest wall pipe in the analysis of a given penetration. In the fourth EPRI validation case, the predicted fluid temperature matched the measured data closely, verifying that the heat transfer model for the entrapped fluid provides an accurate prediction of fluid temperature. In all cases, as the pressure reaches a point where the hoop stress exceeds the pipe material yield strength, plastic straining takes place and there is significant pressure relief in the EPRI test. This is in agreement with a report by Structural Integrity Associates, Inc.[13] who finds that when the pipe exceeds yield, the strain and volume increase will be much higher than in the elastic range providing significant pressure relief. The PTRAP methodology considers elastic stain only, and the predicted pressures with the pipe beyond yield are significantly higher than the measured data. This feature provides an additional degree of conservatism in the GL 96-06 assessment.

PIPE HEATUP & PRESSURIZATION ANALYSIS

The methodology outlined above was used to predict the temperature-pressure response of containment penetration piping listed in Table 1, identified as potentially susceptible to pressurization during the design basis event.

Penetration ID	Line Size	Description	Room #[1]
X-85A	3"	RBCW Piping to Reactor Recirculation. Pumps	19
X-85B	3"	RBCW Piping from Reactor Recirculation. Pumps	19
X-86A	3"	RBCW Piping to Reactor Recirculation. Pumps	19
X-86B	3"	RBCW Piping from Reactor Recirculation. Pumps	19
X-23	4"	RBCCW Piping to Reactor Recirculation. Pumps	18
X-24	4"	RBCCW Piping from Reactor Recirculation. Pumps	18
X-17	6"	RHR Head Spray Piping	18
X-53	8"	RBCW Piping to Drywell Coolers	19
X-54	8"	RBCW Piping from Drywell Coolers	19
X-55	8"	RBCW Piping to Drywell Coolers	19
X-56	8"	RBCW Piping from Drywell Coolers	19

Table 1 : Containment Penetrations

Design Basis Event – Small Break LOCA

The small break accident (SBA) is the design basis event for peak drywell temperature evaluated in Chapter 6 of the Susquehanna Final Safety Analysis Report. The SBA imposes the most severe drywell temperature condition on the penetrations for a postulated LOCA event where the containment function is required to isolate a potential radiological source term. In the current analysis, a 6-hour reactor blowdown is conservatively assumed at maximum drywell superheat conditions of 340 F and 35 psig, consistent with the design basis event. After 6 hours, the reactor blowdown is complete and the Residual Heat Removal system provides decay heat removal and long-term cooling with no credit taken for heat loss to surrounding structures or compartments. No credit is taken for drywell spray operation. Figure 4 provides the SBA drywell temperature history used in the current analysis. All piping outside of primary containment is exposed to post-LOCA secondary-containment temperatures. Figures 5 provides the temperature history for (secondary containment model) Rooms18 and 19 used in the analysis.

[1] Room # refers to the secondary containment model room number that the penetration piping runs through.

A review of the penetration and piping geometry for both units revealed that the geometry between isolation valves was the same for a given penetration with a few exceptions. In a few cases, there was slightly more pipe outside of the drywell on one unit than the other. In these cases, the unit with less piping outside the drywell case was considered since this provided less piping for cooling to the reactor building atmosphere and less fluid mass to serve as a heat sink. In the case of penetration X-23 there was a slight difference in the length of pipe within the drywell, and both units were analyzed to assure that the bounding case was captured.

All of the subject piping is insulated except for that associated with penetrations X-23 and X-24, whose moderate process temperature does not require insulation per the piping design specification. To control piping heatup for penetrations X-23 and X-24 and avoid excessive pressurization with the model assumptions, 1.5 inches of stainless-steel jacketed Koolphen-K anti-sweat insulation is added to the piping model inside the drywell between the penetration and isolation valve. This modification proved very effective in controlling fluid heatup and pipe pressurization. Cases presented in this calculation for penetrations X-23 and X-24 incorporate the Koolphen-K insulation.

Results

The maximum-calculated temperature and pressures for the subject penetrations are presented in Table 2. Results for peak pressure and temperature predicted for each penetration along with the transient time at which the pressure occurred are provided. Also included is the pressure at 24 hours into the transient to provide a sense of the timing for the particular penetration's heatup.

Penetration	P_{max} (psig)	T_{max} (F)	Time (hr)	P(24hrs) (psig)
X-85A	3010	118	84	2047
X-85B	2810	120	66	2270
X-86A	4570	139	64	3613
X-86B	4400	142	52	3860
X-23(U1)	2280[1]	133	10	2078
X-23(U2)	2270[1]	134	11	2094
X-24	2420[1]	135	8	2129
X-17	4600	160	28	4587
X-53	2970	117	330	600
X-54	2570	114	328	695
X-55	3030	114	382	515
X-56	2570	114	318	711

Table 2: Calculated Peak Pressure and Temperature

[1] X-23 and X-24 are analyzed with 1.5" thickness of Koolphen-K insulation added to the penetration piping within the drywell (Currently this pipe is not insulated).

Calculation Uncertainty

The current methodology was developed with the intent of providing a conservative approach to the prediction of thermally-induced pipe pressurization. Best-estimate correlations were employed where practical. However, where the effect of a physical mechanism was uncertain or difficult to quantify, a conservative approach was taken. For example, it is assumed that the thermal radiation source inside the drywell is a blackbody and that the penetration piping is in full view of the radiation source to maximize radiant heating.

Values for input parameters were obtained from reliable published data and design information. The results of sensitivity analyses indicate that the methodology is relatively insensitive to deviations in the input parameters which are within the uncertainty of known values. For instance, a change in the internal radius of the pipe by as much as 15% of the pipe wall thickness produces less than 1% deviation in the maximum pressure. An increase in the external heat transfer coefficient by up to 3 orders of magnitude produces less than 2% increase in the maximum calculated pressure. Mesh and time-step sensitivity studies indicate that the numerical discretization is sufficient to provide calculation accuracy.

Comparison of validation cases to experimental data verified that the current methodology predicts conservative values for pipe pressure. Consideration only of elastic pipe strain incorporates a further degree of conservatism as both theory and experiment demonstrate that even a small amount of plastic straining provides increased pressure relief and reduces the maximum pressure experienced by the piping run.

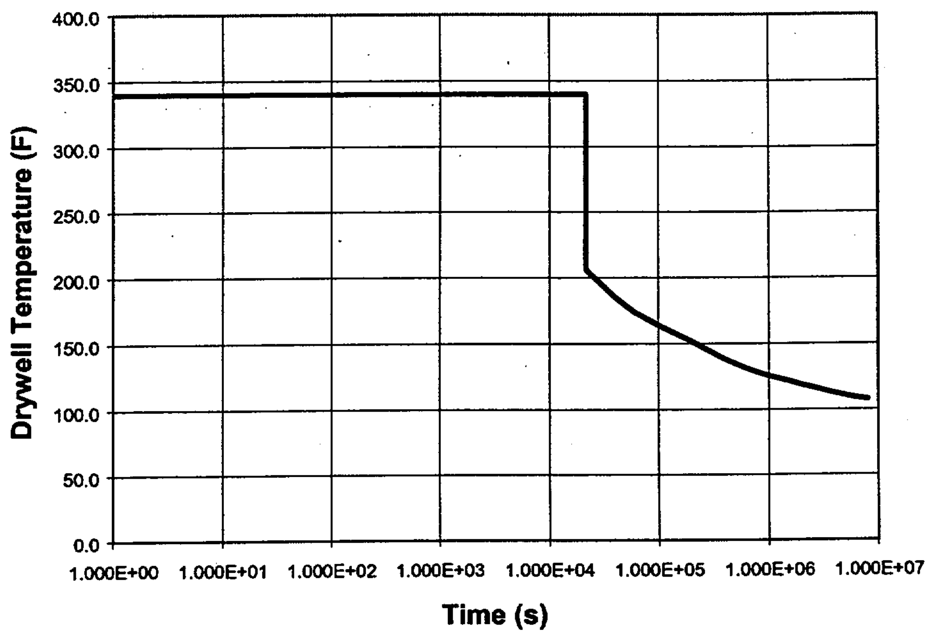

Figure 4: Drywell Temperature History for SBA

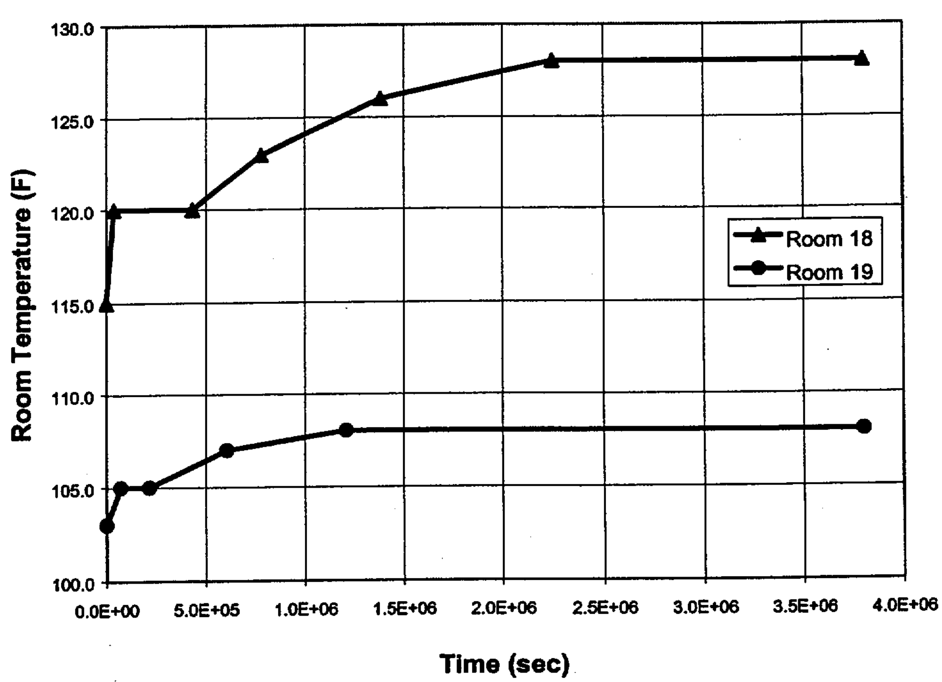

Figure 5: Secondary Containment Temperature - SBA

REFERENCES

(1) Gosselin, S.R., "Response of Isolated Piping to Thermally Induced Overpressurization during a Loss of Coolant Accident (GL96-06)", EPRI TR-108812, Electric Power Research Institute (December 1997).

(2) Press, W.H., Teukolsky, S.A., Vetterling, W.T., & Flannery, B.P., "Numerical Recipes in FORTRAN, *The Art of Scientific Computing*", 2nd Edition, Cambridge University Press, 1992.

(3) Incropera, F.P. & DeWitt, D.P., "Fundamentals of Heat and Mass Transfer", 2nd Edition, Wiley, 1985.

(4) Churchill, S. W., and H. H. S. Chu, "Correlating Equations for Laminar and Turbulent Free Convection from a Vertical Plate," *Int. J. Heat Mass Transfer*, **18**, 1323, 1975.

(5) Churchill, S. W., and H. H. S. Chu, "Correlating Equations for Laminar and Turbulent Free Convection from a Horizontal Cylinder," *Int. J. Heat Mass Transfer*, **18**, 1049, 1975.

(6) Collier, J.G., "Convective Boiling and Condensation", 2nd Edition, McGraw-Hill, 1980.

(7) Rohsenow, W. M., "Heat Transfer and Temperature Distribution in Laminar-Film Condensation," *Trans. ASME*, **78**, 1645-1648, 1956.

(8) Dhir, V., and Lienhard, J., "Laminar Film Condensation on Plane and Axisymmetric Bodies in Nonuniform Gravity", *J. Heat Transfer*, **93**, 97-100, 1971.

(9) Zemansky,M.W. & Dittman, R.H., "Heat and Thermodynamics, *An Intermediate Textbook*", 6th Edition, McGraw-Hill, 1981.

(10) Baumeister, T. "Marks Standard Handbook for Mechanical Engineers", 8th Edition, McGraw Hill, 1978.

(11) Roark, R.J., & Young, W.C., "Formulas for Stress and Strain", 5th Edition, McGraw Hill, 1975.

(12) EC-THYD-1051 Rev.0, "Evaluation of Containment Penetration Piping Heatup and Pressurization during a Postulated Design Basis Accident", PPL Susquehanna LLC, 2001.

(13) Hirschberg, P., "Pressure Due to Fluid Expansion and Leak Tightness of Chilled Water Penetration Piping", Structural Integrity Associates, Inc., File No. PPL-17Q-301, 2000.