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GE Hitachi Nuclear Energy

3901 Castle Hayne Road, Wilmington, NC 28401

NEDO-33347
Revision 0
Class I
DRF 0000-0069-4642
January 2008

Licensing Topical Report

CONTAINMENT OVERPRESSURE CREDIT FOR NET POSITIVE SUCTION HEAD (NPSH)

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TABLE OF CONTENTS

Page

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES — PAGE

LIST OF APPENDIX TABLES — PAGE

LIST OF FIGURES — PAGE

LIST OF APPENDIX FIGURES — PAGE

LIST OF APPENDIX FIGURES PAGE

EXECUTIVE SUMMARY

Historically, both during original licensing and by subsequent license amendments, the U. S. Nuclear Regulatory Commission (NRC) has granted several Boiling Water Reactor (BWR) plant requests to take credit for containment overpressure to ensure adequate Net Positive Suction Head (NPSH) for Emergency Core Cooling System (ECCS) pumps. Over the past decade, several requests have been approved in conjunction with power uprates and suction strainer issues (e.g., NRC Bulletin 96-03 and Generic Letter 97-04). The regulations governing BWR design and operation do not prohibit taking credit for containment overpressure; however, there is no regulatory guidance for use by utilities in requesting credit for containment overpressure.

The BWR Owners Group undertook an effort to develop this licensing topical report at the request of the NRC to provide a standard process for license amendments that request credit for containment overpressure.

This topical report contains a methodology to be utilized by utilities needing to request containment overpressure credit (COP) to provide adequate NPSH for its ECCS pumps. It addresses the following elements:

The risk and safety impacts of taking credit for containment overpressure (it concludes that these risks are generally low with delta Core Damage Frequency (ΔCDF) and delta Large Early Release Frequency (ΔLERF) categorized as very small changes)

ECCS pumps' ability to withstand sustained periods of cavitation without identifiable internal damage.

Those issues generic to the BWR fleet that do not require additional review by the staff for future applicants.

Elements of a license amendment request are included to provide utility applicants with guidance on information to be included. These applications will be submitted only as needed.

This topical report shows that some plants may require some overpressure to avoid temporary cavitation at the 95/95 probability and confidence level. Evaluations presented demonstrate that granting credit for anticipated containment overpressure involves no undue risk to the heath and safety of the public. The same conclusion can be drawn for events other than a Design Basis Accident – Loss of Coolant Accident (DBA - LOCA) such as an Anticipated Transient Without SCRAM (ATWS), Appendix R Fire, or Station Blackout.

ACKNOWLEDGEMENTS

The following individuals contributed significantly to the development, verification and review of this report:

First Name	Last Name	Organization
Tim	Abney	GEH
Francis	Bolger	GEH
James	Dailey	Southern Nuclear Company
Frederick	Emerson	GEH
John	Freeman	Exelon
Suresh	Gupta	GEH
James F.	Harrison	GEH
Guangjun	Li	GEH
Dilip	Rao	GEH
Sang	Rhow	GEH
Alan	Roderick	Nuclear Management Company
Lisa	Schichlein	GEH
Thomas	Simpson	GEH
Mike	Strope	FPL Energy – Duane Arnold Energy Center
Kevin	Whildin	GEH
Alan	Wojchouski	NMC-Monticello BWROG Chair-COP for NPSH Committee
Greg	Wojtowicz	DTE Energy – Fermi

ACRONYMS	
Term	**Definition**
AC	Alternating Current
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARI	Alternate Rod Insertion
ASME	American Society Of Mechanical Engineers
ATWS	Anticipated Transient Without SCRAM
BEP	Best Efficiency Point
BLOCKAGE	Computer Code
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CAFTA	Fault Tree Analysis Computer Code
CDF	Core Damage Frequency
CFR	Code Of Federal Regulations
COP	Containment Overpressure Credit
CS	Core Spray
CST	Condensate Storage Tank
DBA	Design Basis Accident
DBA - LOCA	Design Basis Accident – Loss of Coolant Accident
DW	Drywell
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
GEH	GE Hitachi Nuclear Energy
HEP	Human Error Probability
HLOSS	Computer Code for calculating Head Loss

ACRONYMS	
Term	**Definition**
HRA	Human Reliability Analysis
IE	Initiating Event
ILRT	Integrated Leak Rate Test
INEEL	Idaho National Engineering and Environmental Laboratory
IPEEE	Individual Plant Examination of External Events
La	Containment Leakage Rate (percent/24 hours)
LAR	Licensing Amendment Request
LCO	Limiting Condition of Operation
LERF	Large Early Release Frequency
LLOCA	Large LOCA
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCS	Low-Pressure Core Spray
MNGP	Monticello Nuclear Generating Plant
MSIVC	Main Steam Isolation Valve Closure
MWt	Megawatt Thermal
NPSH	Net Positive Suction Head
$NPSH_a$	NPSH Available
$NPSH_r$	NPSH Required
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ODYN	GEH One-Dimensional Core Transient Model for BWRs
PCPL	Primary Containment Pressure Limit
PRA	Probabilistic Risk Assessment
psi	Pounds Per Square Inch
psia	Pounds Per Square Inch – Absolute
psig	Pounds Per Square Inch – Gauge

ACRONYMS	
Term	**Definition**
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RV	Relief Valve
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SHEX	GEH Containment Analysis Computer Code
SLC	Standby Liquid Control
SPC	Suppression Pool Cooling
SORV	Stuck-open Relief Valve
SRV	Safety Relief Valve
SV	Safety Valve
TRACG	GEH Version of the Transient Reactor Analysis Code
TS	Technical Specification
UHS	Ultimate Heat Sink
VY	Vermont Yankee
WW	Wet Well

1.0 INTRODUCTION

1.1 NEED FOR TOPICAL REPORT

Staff approval of the topical report will provide a standardized, predictable approach for utility applicants to request Containment Overpressure Credit (COP) for Net Positive Suction Head (NPSH), as well as a standardized review and approval process for the U.S. Nuclear Regulatory Commission (NRC). As such, a more predictable licensing process is created which benefits both utilities and the NRC. These applications will be submitted only as needed. This topical report is not intended to provide a generic justification that containment overpressure credit is not needed.

1.2 BACKGROUND / HISTORY OF EVENTS

Historically, both during original licensing and via subsequent license amendments, the NRC has granted requests by several BWR plants for COP to ensure adequate NPSH for Emergency Core Cooling System (ECCS) pumps. Over the past decade, several requests have been approved as a result of power uprates and suction strainer issues (e.g., NRC Bulletin 96-03, Reference 1 and Generic Letter 97-04, Reference 2). The regulations governing Boiling Water Reactor (BWR) design and operation do not prohibit taking credit for containment overpressure, however, there is no regulatory guidance which may be used by utilities in requesting credit for containment overpressure. This has resulted in inconsistent license amendment requests for COP and concerns raised by the Advisory Committee on Reactor Safeguards (ACRS) that there are no clear acceptance criteria for NRC staff approval. NRC senior management contacted the BWR Owners Group (BWROG) and requested a meeting to discuss the issue. Follow-up discussions were held with the NRC staff.

As a result of these discussions, the BWROG is proposing this licensing topical report to address the need for COP. This topical report addresses the following elements:

- The risks and safety impacts of taking credit for containment overpressure
- ECCS pumps ability to withstand sustained periods of cavitation without identifiable internal damage.
- Those issues generic to the BWR fleet that do not require additional review by the NRC staff for future applicants.

Issues generic to BWR fleet not expected to be repeated in future applications for COP are addressed in Sections 2.1, 2.2, 3.3, 5.1, Appendix B and Appendix C. Sections 6.0 and 7.0 give the generic acceptance criteria and required elements for licensing submittal requesting COP applicable to future BWR fleet requests for COP.

1.3 CONTENT / USE OF TOPICAL REPORT

This topical report addresses the following key areas:

1. Available NPSH evaluation for Design Basis Accident – Loss of Coolant Accident (DBA-LOCA) (Section 3.0)
2. The proposed licensing basis methodology for NPSH evaluation (Section 6.0)

3. The safety basis for requesting COP (Sections 5.1 and 5.3)
4. Elements of a license amendment request for COP (Section 7.0).

This topical report is to be used by licensees as an acceptable approach when applying for or increasing COP to ensure net positive suction head for ECCS pumps.

1.4 DEVELOPMENT OF TOPICAL REPORT

This topical report was developed by systematically addressing safety issues associated with taking credit for containment overpressure. A review of previous licensee submittals was also performed to understand previous approaches taken and issues addressed to ensure that all potential safety issues were addressed.

1.5 REGULATORY BASIS OF TOPICAL REPORT

The regulations governing nuclear plant design and operation do not prohibit taking credit for containment overpressure. The NRC has allowed the utilization of COP in NPSH analyses for both pressurized water reactors (PWRs) and BWRs. Thus, under the current regulations, the NRC has the authority to approve COP requests per Code of Federal Regulations (CFR) 10 CFR 50.92, Issuance of Amendment, for an analysis that indicates no significant hazards. To credit containment overpressure in determining available NPSH, licensees must provide reasonable assurance that sufficient containment pressure will be available when required during postulated events. This topical report provides the reasonable assurance from a generic perspective and identifies those issues that must be addressed on a plant-specific basis.

1.6 SCOPE AND LIMITATIONS OF TOPICAL REPORT

This topical report contains a methodology to be utilized by utilities needing to request COP to provide adequate NPSH for its ECCS pumps, as part of Extended Power Uprate (EPU) or other applications. The NRC approved licensing basis methodology for containment pressure and temperature response is described in Appendix G of Reference 3. This methodology, as applied for NPSH calculations, is described in Section 3.1.1. The licensing basis methodology forms the basis for requested COP. Additional statistical analysis (Section 3.1.2) provides a demonstration of additional containment overpressure margin.

The BWROG Containment Overpressure Credit Committee prepared this Licensing Topical Report with support from GEH.

2.0 OVERVIEW OF NPSH EVALUATION

When a liquid flows into a region where its pressure is reduced to vapor pressure, it boils and vapor pockets develop in the liquid. The vapor bubbles are carried along with the liquid until a region of higher pressure is reached, where they suddenly collapse. This phenomenon is called cavitation. If cavitation occurs in a pump, it causes undesirable conditions, such as lowered efficiency and potential damage to flow passages, noise and vibration. To prevent cavitation from occurring, a positive pressure at the pump inlet is required, which is called Net Positive Suction Head (NPSH).

The pump manufacturer specifies a required minimum value of NPSH to ensure that only minimal acceptable cavitation will occur assuming the pump cavitates at this level for its entire life cycle. For a well-defined fluid system consisting of a pump, it is ensured that adequate NPSH is available (i.e., the available NPSH ($NPSH_a$) is greater than or equal to the required NPSH ($NPSH_r$)).

Section 3.0 discusses the technical basis for evaluating the available NPSH for the DBA-LOCA. A discussion of special events such as Anticipated Transient Without SCRAM (ATWS), Station Blackout (SBO) and Appendix R is provided in Section 4.0. Other evaluations associated with NPSH are described in Section 5.0. Section 6.0 provides a description of the methodology proposed for the evaluation of the available NPSH. Section 7.0 describes elements of license amendment request for COP. Appendices A and B present the results of applying the methodology to an example plant for the DBA-LOCA. Appendix C describes the alternate methods to containment pressure credit. Appendix D describes the basis for containment overpressure credit.

2.1 REQUIRED NPSH

$NPSH_r$ is defined as the NPSH at which the pump total head (first stage head in multi-stage pumps) has decreased by three percent (3%) due to low suction head and resultant cavitations within the pump.

Pump vendor or manufacturer determined required pressure is expressed in feet of liquid for operation for a given pump or class of pumps. They use the Hydraulic Institute Standard (Reference 21 and Reference 22) criteria for establishing the $NPSH_r$ as the point where cavitation drops dynamic head by 3% as measured in the test rig. This number is typically shown on the pump performance curve supplied by the pump manufacturer.

Acceptable pump operation is defined as:

$NPSH_a - NPSH_r \geq 0$ feet

2.2 AVAILABLE NPSH

The $NPSH_a$ is the total suction head of liquid (in feet) available above vapor pressure at the pump inlet, and this value for pumps drawing from the suppression pool is calculated as:

$$NPSH_a = (P_{ww} - P_v) \times 144/\rho_w + H_{pool} - H_{pump} - H_{loss} \quad (1)$$

Where:

$NPSH_a$	Available NPSH for pump (ft)
P_{ww}	Wetwell airspace pressure (psia)
P_v	Saturation vapor pressure at suppression pool temperature (psia)
ρ_w	Density of suppression pool water (lbm/ft^3)
H_{pool}	Elevation of suppression pool surface above the pump suction (ft)
H_{pump}	Elevation of pump suction (ft)
H_{loss}	Suction strainer and suction line losses from suppression pool to pump (ft)

Typically, the last term (H_{loss}) is calculated separately from the containment response analysis that provides the value of the first two terms, and the third term (H_{pump}) is a constant value for a given pump location. In view of this, the $NPSH_a$ may be split into two parts as defined below:

$$NPSH_a = H_{ww} + H_{pl} \qquad (2)$$

Where:

$$H_{ww} = (P_{ww} - P_v) \times 144/\rho_w$$

$$H_{pl} = + H_{pool} - H_{pump} - H_{loss}$$

During postulated events (e.g., DBA-LOCA), the ECCS and Residual Heat Removal (RHR) pumps draw water from the suppression pool, and the events are analyzed to ensure that the $NPSH_a$ is greater than the $NPSH_r$ for these pumps. Equations (1) and (2) indicate that the $NPSH_a$ is both event-dependent and time-dependent. For instance, as the suppression pool temperature increases with time during the DBA-LOCA, the saturation pressure increases, and therefore the $NPSH_a$ will decrease. The wetwell pressure above the suppression pool also increases as the containment temperature increases, thus providing containment overpressure. In addition, for the LOCA the $NPSH_a$ will increase with an increase in the suppression pool surface level due to the discharge of the reactor vessel inventory to the containment. The last term in Equation (1) (suction strainer and line loss) for a given flow rate is dependent on the debris buildup assumption made on the blockage condition of ECCS suction strainers during the DBA-LOCA.

The approach is to first identify the limiting single failure for $NPSH_a$ cases separately for the short term and long term scenarios and then to calculate the containment responses deterministically, using licensing bases input values. The containment analysis provides time histories for suppression pool temperature, wetwell airspace pressure and suppression pool water level. The suppression pool water density and saturated vapor pressure are derived from the suppression pool temperatures time history. Therefore, the containment analysis contains all the information needed to determine $NPSH_a$, except for the elevation head above the pump inlet and the suction strainer and suction line head losses. The suction strainer and suction line losses are unique to the plant design and its licensing bases. The containment responses are used to determine the suction strainer and suction line losses at specific time steps to calculate H_{loss} and the corresponding $NPSH_a$ time history.

The analysis, if necessary, can also determine how much containment overpressure credit is required to have $NPSH_a = NPSH_r$. If COP is required, then an additional statistical containment analysis will be performed. The additional statistical analysis will provide expected values of the containment overpressure needed when realistic input values are used. This statistical containment analysis uses the same limiting single failure and scenarios as the deterministic analysis, but uses certain inputs each based on randomly generated factor applied to their input exceedance probability. Monte Carlo statistical containment analysis utilizing 59 independent analysis cases is performed. The resultant time history, time step distribution, maximum value, mean value and minimum value of the wetwell air space pressure, suppression pool water temperature and H_{ww} are determined. It needs to be noted that the minimum value of H_{ww} for each time step does not necessary correspond to the lowest wetwell air space pressure or the highest suppression pool water temperature or the lowest suppression pool water level. The cumulative effect of all of the input factors is used in the determination process. The simplest and most conservative approach is to utilize the minimum H_{ww} with the highest suppression pool water temperature when calculating $NPSH_a$ and comparing the resultant required containment overpressure to the minimum wetwell air space pressure. This would be representative of a statistical 100% confidence interval. If desired a more complicated and less conservative approach that establishes a 95/95 probability and confidence level can be employed.

2.3 REQUIRED NPSH FOR ECCS / RHR PUMPS

As discussed in Section 2.1, $NPSH_r$ is defined by industry standards as the NPSH at which the pump total head has decreased by three percent (3%) due to low suction head and resultant cavitation within the pump. Pump manufacturers typically show the $NPSH_r$ curve on the pump performance curve. These values establish acceptability for long-term pump operation, however they are overly conservative for analysis of the infrequent events with limited duration in which COP may be required. Therefore, establishing $NPSH_r$ values commensurate with the minimum hydraulic performance requirements for each pump and the time duration over which an amount of $NPSH_a$ exists can reduce excess conservatism.

Experience has shown that vendor curves originally supplied with the ECCS pumps may represent a more stringent one percent (1%) total head decrease instead of the Hydraulic Institute Standard 3%. The pump testing conducted at the time of pump delivery was to demonstrate compliance to the purchase specifications and a standard 3% $NPSH_r$ test was not required (Reference 4).

When evaluating if COP is required, the most realistic $NPSH_r$ curve available should be used, or at a minimum it should be identified that a more conservative (i.e., 1%) $NPSH_r$ curve was used.

The $NPSH_r$ values are plant-specific based on the manufacturer and model of the ECCS pumps.

2.4 SUMMARY AND CONCLUSIONS

To assure continuous pump operation with minimal acceptable pump cavitation, the net positive suction head available needs to be equal to or greater than the net positive suction head required. Net positive suction head required is determined by test as the point where cavitation drops the total dynamic head by 3%. The net positive suction head available is the total suction head of liquid available above the vapor pressure at the pump inlet.

It is expected that the licensee will demonstrate that $NPSH_a - NPSH_r \geq 0$; when considering design basis accidents including considerations for additional single failures.

It is expected that the licensee will demonstrate that $NPSH_a - NPSH_r \geq 0$; for special events.

3.0 AVAILABLE NPSH EVALUATION FOR DBA-LOCA

3.1 CONTAINMENT PRESSURE AND TEMPERATURE RESPONSE

The DBA-LOCA, a double-ended break of a recirculation suction line, is analyzed to confirm that adequate NPSH is available for pumps that draw water from the suppression pool. For the DBA-LOCA, only low-pressure ECCS is credited. The low-pressure ECCS consists of core spray (CS) and low-pressure coolant injection (LPCI)/residual heat removal (RHR) systems. The DBA-LOCA is the bounding event for LOCAs in terms of the $NPSH_a$, since the DBA-LOCA will result in higher peak suppression pool temperature and a higher degree of debris blockage for ECCS suction strainers than other events. Since the DBA-LOCA is a design basis event, it is analyzed with a single failure assumption that would result in the worst accident scenario.

The DBA-LOCA is analyzed for two time domains: 1) short-term (typically the first 10 minutes into the event) with no credit for possible operator action, and 2) long-term after operator action for containment cooling. For both time domains, accident scenarios that include a single failure assumption are developed such that the suppression pool temperature is maximized while the wetwell pressure is minimized, so that the resulting $NPSH_a$ is minimized. The accident scenarios to be analyzed are plant-specific.

A typical BWR plant that may require COP was selected to be the subject of the analysis and to demonstrate the methodology. For the Monticello Nuclear Generating Plant (MNGP), which was analyzed as an example plant (see Appendix A), the short-term scenario assumes that all of the LPCI flow is directed into the drywell due to failure of the LPCI loop selection logic, while the long-term scenario assumes failure of a diesel generator that results in the minimum containment cooling capability. Based on plant design, the bounding single failure for the short term and the long-term scenarios can be plant-specific and the plant's licensing bases bounding single failure for each scenario needs to be applied.

Note that during the short-term DBA-LOCA, runout flows occur for ECCS pumps as the vessel is depressurized. On the other hand, the operator can throttle the flow from ECCS pumps at the time of containment cooling initiation. As a result, the $NPSH_r$ (which depends on the pump flow rate) can be different for the same pump between the short-term and long-term DBA-LOCA.

Once the accident scenarios are defined, the DBA-LOCA can be analyzed in two ways:

Deterministic Approach: In this approach, a traditional conservative analysis is performed, using conservative assumptions and input values, such as bounding values for containment initial conditions, such that the resulting pool temperature response is higher and the pressure response is lower. This approach will give a conservative assessment of $NPSH_a$.

Statistical Approach: This approach takes credit for variabilities in the analysis input values. The order statistics method is recommended in Reference 5 and Reference 7. The input variables are defined statistically and combined through a Monte Carlo process, in which 59 random draws are made from the corresponding probability distributions. Containment pressure and temperature time-histories are calculated for the 59 cases. This approach allows for calculating more realistic $NPSH_a$ values, which can be used to quantify the conservatism in the deterministic analysis.

Deterministic Analysis with Conservative Assumptions

The deterministic containment analysis for NPSH evaluations uses conservative input values. For instance, the input values for containment initial conditions, such as suppression pool temperature, pool volume, etc., are bounding values, since they are assumed to be Limiting Condition of Operation (LCO) values as specified in the Technical Specification. If not specified in the Technical Specification, the maximum or minimum expected values are used, depending upon which direction is conservative.

Specifically, the following conservative input assumptions are used:

1. The reactor initial thermal power is at 102% of rated licensed power.
2. The decay heat value after reactor SCRAM is a nominal value plus 2-sigma uncertainty.
3. The suppression pool initial temperature is at its maximum normal operating Technical Specification value.
4. The suppression pool initial volume is at its minimum Technical Specification value.
5. The RHR service water temperature is at its maximum licensing basis value.
6. The analysis uses a conservatively determined minimum value of the RHR heat exchanger heat transfer factor.
7. The containment (drywell/wetwell) initial pressure is at its minimum value.
8. The initial relative humidity in both the drywell and the wetwell is 100%.
9. The containment leakage rate is its maximum allowed value specified in the Technical Specification.
10. The portion of feedwater inventory initially above 212°F flows into the reactor vessel after absorbing heat from the feedwater pipe metal as it flows toward the vessel.
11. [[

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12. The operator initiates suppression pool cooling in containment spray mode (rather than direct pool cooling) at a plant-specific time (typically 10 minutes into the event), and the spray droplets are in thermal equilibrium with the containment airspace before falling on the bottom of drywell or the suppression pool.

Using the above input assumptions, the containment response to the DBA-LOCA is analyzed for the short-term and long-term time domains. The following output parameters are used in the evaluation of the available NPSH, as Equation 1 (See Section 2.2) indicates:

- Wetwell Pressure (P_{ww})
- Suppression Pool Temperature to obtain saturation vapor pressure (P_v) and Density of Suppression Pool water (ρ_w)
- Elevation of suppression pool surface (H_{pool})

For the deterministic approach, a single value for each of the above three parameters is calculated as a function of time. Based on these values, H_{ww} (as defined in Equation (2)) is calculated, and $NPSH_a$ is calculated as a function of time by adding H_{pl} to H_{ww} (see Equation (2)).

3.1.1 Containment Response Sensitivity to Input Parameters

Using the GEH containment analysis computer program SHEX, a sensitivity study was performed for the example plant (MNGP) to evaluate the effect of conservatisms on peak suppression pool temperature and wetwell pressure at the time of peak pool temperature during the DBA-LOCA. The purpose of the study was to identify the significant input parameters that impact the containment response. The sensitivity study was limited to the input parameters for which uncertainties can be statistically defined.

Based on review of historical MNGP data, the service water temperature varies from 35°F to 90°F with its median value below 50°F. The deterministic approach would use the maximum value of 90°F for the service water temperature in the DBA-LOCA analysis. Review of the Table 3-1 sensitivity results indicates that the peak suppression pool temperature with a reduced service water temperature would be significantly lower, compared with the result obtained with the maximum service water temperature. Likewise, the use of more realistic values for initial containment pressure would result in a higher containment pressure response. Thus, the deterministic approach with conservative assumptions as listed in Section 3.1 will result in a higher suppression pool temperature combined with a lower wetwell pressure, thus providing a conservative extreme lower limit value for the available NPSH.

Table 3-1 summarizes the results of the sensitivity study.

Table 3-1
Sensitivity of DBA-LOCA Containment Response to Input Parameters

Key Input Parameter	Variation	Primary Output Parameter of Concern	Uncertainty Effect
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			]]

* The nominal + 2 Sigma decay heat values were reduced by 6% to approximate a reduction of 2 Sigma on the decay heat.

3.1.2 Statistical Approach with Realistic Assumptions

For the deterministic approach described in Section 3.1, either the maximum or the minimum value is used for each input parameter, depending upon which direction is conservative. On a realistic basis, all the input parameters will not be at their extreme (maximum or minimum) values at the same time. For the statistical approach with realistic assumptions, input parameters that can be statistically defined are selected first, based on available information. Typically, the following input parameters can be statistically defined:

For core power generation during the event:

- Initial reactor power - The uncertainty for this parameter stems from uncertainties in measurement data used in determining the reactor power.
- Decay heat value after reactor SCRAM – The uncertainty is determined consistent with the methodology for calculating decay heat.

The following nine parameters may be measured periodically at power plants, and the measurement data will exhibit a probability distribution for each parameter:

- Initial suppression pool temperature
- Service water (ultimate heat sink) temperature
- RHR heat exchanger heat removal capability
- Initial suppression pool volume

- Initial drywell temperature
- Initial drywell pressure
- Initial wetwell pressure
- Initial drywell and wetwell airspace relative humidity
- Initial containment leakage rate – Containment leakage tests are performed at power plants (10CFR Appendix J to Part 50 test) to ensure that the containment leakage rate is less than the value specified in the Technical Specification. Individual power plants provide a very limited data set beyond the Technical Specification allowed leakage rate. Because of lack of plant specific data, a compilation of industry-wide containment leakage data as provided in an Electric Power Research Institute (EPRI) report (Reference 6) is used. The EPRI report assesses, based on the compilation of industry-wide containment leakage events, the likelihood of pre-existing containment leakages, and summarizes the resulting probabilities as a function of pre-existing leakage sizes.

Once the input parameters are defined, variations in the input parameters are combined through a Monte Carlo process, in which random draws are made from the corresponding probability distributions in order to determine input values. A calculation of the containment response with one set of randomly drawn input values represents one trial in the statistical process. At least 59 trials (calculations) are made to obtain statistically meaningful results at 95%-probability and 95%-confidence (95/95) level consistent with the order statistics method (References 5 and 7). Appendix A shows the application of this statistical approach to the example plant.

As mentioned previously, the following output parameters are used in the evaluation of the $NPSH_a$:

- Wetwell airspace pressure (P_{ww})
- Suppression Pool Temperature to obtain saturation vapor pressure (P_v) and Density of Suppression Pool water (ρ_w)
- Elevation of suppression pool surface (H_{pool})

Based on the values of the above output parameters, the value of H_{ww} is calculated as a function of time for each of the multiple 59 trials (calculations). From the set of 59 time-histories, the minimum values of H_{ww} are obtained as a function of time, and the resulting minimum values are used as 95/95 values. In summary, for the deterministic approach with conservative assumptions, the time history of H_{ww} is used directly in the $NPSH_a$ calculation. For the statistical approach with realistic assumptions, the time history of minimum H_{ww} determined from at least 59 calculations is used as its 95/95 value.

Both the deterministic approach and the statistical approach are applied to the example plant (MNGP) for the DBA-LOCA described above. The results are presented in Appendix A. These two approaches are also applied for a hypothetical case where containment overpressure is not credited, and the results are presented in Appendix B.

3.2 SUCTION STRAINER AND SUCTION LINE HEAD LOSSES FROM SUPPRESSION POOL TO PUMP INLET

BWR plants are expected to retain existing licensing basis strainer debris analysis loading methods.

ECCS suction strainer blockage during a DBA-LOCA was identified as a concern during the mid-1990s. This concern was communicated to the nuclear industry via the issuance of NRC Bulletin 96-03 (Reference 1). As a result of the identified concern, the BWROG prepared and issued the GE topical report NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage". The NEDO-32686 (Reference 8) provided guidance to the BWR licensees to evaluate ECCS pump suction strainer blockage as a result of the effects of a postulated LOCA. NEDO-32686 was intended as a guide; thus, there are numerous plants–specific considerations that dictate unique evaluations and resolutions by each plant.

Each BWR licensee has evaluated the effects of LOCA-generated debris on the ECCS suction strainers and has incorporated the methods and results into their licensing basis.

When calculating the suction strainer head losses, existing licensing basis methods need to be employed. Any changes to ECCS suction strainer head loss determination from the sites' existing licensing basis methods would need to be identified and justified.

When determining suction strainer and suction line head losses, existing evaluations should be reviewed to identify unnecessary conservatisms in the evaluation inputs that could be reduced, rather than relying on accident pressure to assure adequate NPSH.

3.3 UTILITY SURVEY OF NPSH EVALUATION

3.3.1 Review Performed

Plants participating in the BWROG Containment Overpressure Credit Committee submitted their low-pressure ECCS pump NPSH calculations for review. Participating plant $NPSH_a$ calculations, including those who previously submitted applications for COP, were reviewed for the typical conservatism in the calculations.

Based on review of the Core Spray NPSH calculations that were submitted for review, two general groups of conservatisms were identified: the containment analysis group and the piping analysis group. The containment analysis provided the boundary conditions for the piping system pressure drop. The containment analysis parameters where conservatism is introduced can be summarized:

- Maximum Suppression Pool Temp
- Maximum Service Water Temp
- Maximum Decay Heat
- Maximum Containment Spray
- Maximum Reactor Coolant System (RCS) inventory release
- Minimum Dry Well (DW) Pressure

The second group is for the piping system pressure drop that can be summarized as:

- Scenario Selection
- Strainer Plugging
- Pump Flows
- Piping length
- Elbow, Tee and Valve Losses
- Piping Aging & Friction
- Minimum level in pool
- Water Properties
- Manual calculations
- Manual Hydraulic (Head loss) calculation methodology
- Piping Systems modeling in software used for Hydraulic (Head loss) calculations

3.3.2 Analysis of Utility Piping System Conservatism

Utility piping system calculations always include conservatism scenario selection related to both the containment analysis group and the piping analysis group. For addressing scenario selection within the DBA-LOCA, 10CFR50 Appendix K requires that worst combination of break size, break location and single failure be addressed. While a double-ended guillotine break of the recirculation suction line and failure of a diesel generator may be the worst challenge to fuel thermal limits in 10CFR50.46 criteria, it may not be the most challenging for the NPSH criteria. For NPSH, the most challenging scenario might include no failures or a single failure that maximizes ECCS flow rate through the suction piping of interest. If the suction piping is common to more than one pump, any single failure that maximizes flow through the common suction pipe can be used to minimize NPSH by means of increased frictional pressure drop due to higher flow velocities. For BWR 3-4 plants with LPCI loop selection logic, a single failure of the loop selection logic allows all the LPCI pumps to inject to the broken loop offering little resistance while the remaining sprays are also running at maximum capacity. This scenario minimizes $NPSH_a$ although it may not be limiting for the fuel response, or even the containment response.

3.3.3 Survey Results

Utility piping system calculations include conservatism in some, but not all, the other types outlined above. Considering the results of the survey and the fact that conservatism always exists in the containment analysis and to varying degrees in the system piping analysis, there is always significant conservatism in the $NPSH_a$ analyses. While there is not consistency of margin among the entire fleet for a direct Core Spray system comparison, for a given combination of Nuclear Steam Supply System (NSSS), containment and pump designs the results have consistently shown that COP is justified. The Core Spray results typify the conservatisms in ECCS NPSH analyses.

In summary, it can be concluded that most plants do include elements of conservatism in their NPSH calculations. However, elimination of these conservatisms alone may not be adequate to eliminate the need for COP.

3.3.4 Expectation of Significant Differences

There is a wide variety of combinations given the BWR 2-6 Nuclear Steam Supply System (NSSS) products and the Mark I, II, and III containment designs. In addition, each plant has pump inlet piping sizes and suction strainers installed of a typically unique design. Pump technology was advanced with the submerged well pump design that inherently provided adequate $NPSH_a$ simply due to the height of the pump inlet and impeller assembly. Changes to the plant-licensing basis that increase long-term suppression pool temperatures can reduce NPSH margin. However, the increased temperature in the pool will typically also increase the available overpressure. Each plant has unique RHR capability that changes the pool temperature for a given combination of NSSS and containment. Therefore, to resolve the differences requires an understanding of the total combination of conservatisms between the containment response and piping system analysis conservatisms. As expected, the newer designs for NSSS and containment fare better in not requiring as much, if any COP. As expected, the older combinations will require more COP than their newer counterparts.

3.3.5 Summary of Methodology to be Used By Licensees

The methodology to be used by licensees was outlined in Sections 2.1, 2.2 and 2.3. This methodology is consistent with results of the utility survey. In practice, hand calculations, computer based calculations or a combination of both can be used according to standard engineering practice typical of Reference 9. Where greater accuracy is available, e.g. clean suction strainer loss coefficients, or other pressure loss data, it may be used as a basis for developing the terms for suction strainer and/or line losses.

3.4 SUMMARY AND CONCLUSIONS OF AVAILABLE NPSH FOR DBA LOCA

The DBA-LOCA shall be analyzed for two time domains: 1) short-term (typically the first 10 minutes into the event) with no credit for possible operator action, and 2) long-term after operator action for containment cooling. For both time domains, accident scenarios that include a single failure assumption are developed such that the suppression pool temperature is maximized, while the wetwell pressure is minimized, so that the resulting $NPSH_a$ is minimized.

Once the limiting accident scenarios are defined, the DBA-LOCA can be analyzed in one of two ways:

- Deterministic Approach: Inputs and assumptions as specified in Section 3.1
- Statistical Approach: Inputs and assumptions as specified in Section 3.1.2

The following output parameters from either approach are used in the evaluation of the $NPSH_a$:

- Wetwell airspace pressure (P_{ww})
- Suppression Pool Temperature to obtain saturation vapor pressure (P_v) and Density of Suppression Pool water (ρ_w) water properties for $NPSH_a$ for use in calculating H_{loss}.
- Elevation of suppression pool surface (H_{pool})

For the deterministic approach with conservative assumptions, the time history of H_{ww} is used directly in the $NPSH_a$ calculation as specified in Section 2.2. For the statistical approach with realistic assumptions, the time history of minimum H_{ww} determined from at least 59 calculations is used as its 95/95 value.

When calculating the suction strainer head losses, existing licensing basis methods need to be employed consistent with Reference 8. Any changes to ECCS suction strainer head loss determination from the sites' existing licensing bases methods would need to be identified and justified. When determining suction line head losses, existing evaluations should be reviewed to identify conservatisms in the evaluation inputs that could be reduced, rather than relying on accident pressure to assure adequate NPSH.

Utility piping $NPSH_a$ calculations include conservatism in some, but not all the types outlined specified in Section 3.3.1.

It is expected that the piping system calculations that contribute to the determination of $NPSH_a$ be generally conservative in nature, while eliminating excessive conservatism.

It is expected that the deterministic approach be utilized for applications requesting credit for accident generated containment pressure.

For applications that are requesting increases in containment overpressure credit or requesting credit for the first time it is expected that analysis utilizing the statistical approach be provided to quantify the expected margins to the realistic $NPSH_a$ values.

4.0 NPSH EVALUATION FOR SPECIAL EVENTS (ATWS, SBO, APPENDIX R)

The adequacy of available NPSH is evaluated using deterministic methodologies for Special Events. Many of the considerations previously described for DBA/LOCA (i.e. containment pressure and temperature response, suction strainer, suction line head losses, and pump NPSH capability) are equally applicable to Special Events.

Safety analyses associated with the DBA-LOCA are predicated on the occurrence of the LOCA, itself a limiting fault, and an additional postulated single active failure in the safety-related primary or supporting systems credited to provide ECCS functions or to mitigate the effects of a radiological release. With few exceptions, non-safety grade components are typically not relied upon for accident mitigation.

In contrast, special events consist of expected transients combined with an additional specified failure characteristic of the event that is beyond the design basis of the plant (e.g. an anticipated transient with the assumed failure of the primary reactor SCRAM system). For these beyond design basis events, other equipment failures are not assumed unless they occur as a consequential failure of the event itself. In addition, the full complement of plant equipment not impacted by the event (i.e. safety and non-safety related) is assumed to be available for mitigation of special event.

4.1 CONTAINMENT PRESSURE AND TEMPERATURE RESPONSE

As with DBA/LOCA, parameters defined in Section 3.1 are important to containment response and $NPSH_a$. In deterministic analysis, these parameters are often set conservatively even though nominal operating and equipment input parameters are considered acceptable for Special Events.

For purposes of determination of $NPSH_a$, it is acceptable to either (1) set critical containment input to nominal values representative of the population of these parameters for deterministic analysis, or (2) demonstrate conservative NPSH margin relative to the mean of a statistical analysis done in a manner similar to that described in Section 3.1.2.

Note that the analysis must consider the full complement of plant equipment (i.e. safety and non-safety related) when evaluating the consequences of the Special Events. If credit is taken for COP, available equipment (i.e. drywell coolers or sprays) must be included in the analysis since the amount of available COP may be reduced.

4.2 SUCTION STRAINER AND SUCTION LINE HEAD LOSSES FROM SUPPRESSION POOL TO PUMP INLET

The requirements of Reference 8 for the DBA/LOCA are not applicable to Special Events since there is no debris loading. However, for plants with Safety Valves (SVs) or Relief Valves (RVs) discharging to the drywell, the plant specific submittal shall address the potential for a drywell SV or RV discharge in each special event and potential for significant debris loading.

4.3 PROPOSED METHODS FOR EVALUATION

The $NPSH_a$ determinations will be completed on a plant-specific basis. It is expected that the deterministic approach utilizing nominal input values will be used to calculate $NPSH_a$ for special events. Should this approach not satisfactorily show that $NPSH_a - NPSH_r \geq 0$; then the statistical approach utilizing the mean output values will be used to show the expected realistic response to the event.

4.3.1 ATWS

The acceptance criteria described in Section 4.1 applies for ATWS (10CFR50.62 - Reference 23, NUREG-0460 - Reference 24) event. Nominal values can vary up to the mean. If the acceptance criteria are met, the deterministic analysis is adequate. If not, the licensee should perform the realistic analysis.

Since an ATWS peak containment pressure does not exceed the peak containment pressure generated during the DBA-LOCA, a challenge to primary containment integrity and a consequential failure of the containment are not credible. Considering the probability of the initiating event, the failure of the primary SCRAM logic and or equipment, and the independent failure of the containment, the probability of experiencing an ATWS without containment integrity is negligibly small, and there is a reasonable expectation that the accompanying necessary containment overpressure response will be present. Notwithstanding this expectation, given the availability of a full complement of ECCS equipment and credit for the margin represented by realistic equipment and heat sink performance, it may be possible to demonstrate that adequate suppression pool sub-cooling can be maintained even without credit for containment overpressure. In other words, the method of evaluation proposed for the DBA-LOCA is directly applicable to the evaluation of the ATWS, except that realistic inputs and assumptions are used.

Note that if the realistic analyses demonstrate that if the realistic analyses demonstrate that credit for containment overpressure protection is required to ensure adequate ECCS pump NPSH, special consideration should be given to potentially non-conservative modeling assumptions such as the assumed loss of drywell cooling and the potential for operator action to initiate drywell sprays. The post-accident operation of either the drywell coolers or the sprays will tend to reduce containment pressure and hence the available NPSH. If in the interest of preserving containment overpressure the plant elects to procedurally turn off drywell cooling or prohibit the operation of the sprays, then special consideration must be given to the limiting operating temperatures of the Safety Relief Valves (SRVs) and the under-vessel neutron monitoring instrumentation (cables) as these are functions that are credited in the ATWS response.

4.3.2 Station Blackout

The acceptance criteria described in Section 4.1 applies for SBO (10CFR50.63 - Reference 25, NUREG-1776 - Reference 26) event. The SBO is similar to the ATWS in that the single failure criterion is not applied for design and the event itself does not exceed the LOCA challenge to primary containment integrity. Hence, the consequential failure of the containment is not credible. Based on the above, the method of evaluation proposed for the DBA-LOCA is directly applicable to the evaluation of the SBO, except that realistic inputs and assumptions are used.

Similar to the ATWS, if the realistic analyses demonstrate that credit for containment overpressure protection is required to ensure adequate ECCS pump NPSH, special consideration should be given to the potential impact on containment overpressure due to post-accident operation of the drywell cooling and drywell sprays. Conversely, the impact of intentionally inhibiting these functions to maintaining a higher post-accident containment temperature and pressure should be evaluated for the potential adverse impact on credited functions such as SRV operation.

4.3.3 Appendix R - Fire

The acceptance criteria described in Section 4.1 applies for 10CFR50 Appendix R (Reference 27) - Fire - 10CFR50.48 (Reference 28) event. It is proposed that the containment response to the Appendix R and alternate (dedicated) shutdown scenarios be evaluated based on the designed complement of containment cooling sub-systems - i.e., a division free of fire damage for the III.G fire, or limited complement associated with the alternate shutdown systems designed to respond to the III.L fire - deterministically with nominal input values or statistical utilizing the mean output values. It is expected that the results of a realistic containment analysis will demonstrate that the need for COP is either eliminated or significantly reduced.

In the event that the realistic analysis cannot demonstrate that adequate ECCS pump NPSH exists without COP, additional design considerations apply. Specifically, whereas there are no containment failure modes that are postulated to occur concurrently with or as a consequence of the ATWS and SBO events, plant fires are postulated to cause spurious actuations of plant equipment - such as primary containment isolation valves - that could potentially challenge the integrity of the primary containment. Specific examples include the torus hardened vent isolation valves, the containment purge valves, and the torus-to-reactor building vacuum breakers. Credible fire-induced failures that could result in a loss of containment overpressure should be identified and eliminated.

Similar to the ATWS and SBO events, if the realistic analyses demonstrate that credit for containment overpressure protection is required to ensure adequate ECCS pump NPSH, special consideration should be given to the potential impact on containment overpressure due to post-accident operation of the drywell cooling and drywell sprays. Conversely, the impact of intentionally inhibiting these functions on maintaining a higher post-accident containment temperature and pressure should be evaluated for the potential adverse impact on other credited on credited functions such as SRV operation.

4.4 RELAXATION OF CONSERVATISMS FOR SPECIAL EVENTS

The following assumptions describe typical conservatisms present in the approved analysis codes methods that could be adjusted to demonstrate available margin in the pressure and temperature time-history calculations in more realistic analyses. Assumptions that reduce the suppression pool temperature would also reduce the wetwell airspace pressure, which would partially offset the benefit of the lower pool temperature.

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Relaxation of typical input values used in LOCA licensing analyses, will produce more realistic results for the special events (ATWS, SBO, and Appendix R). The following relaxations may be considered. Assumptions that reduce the suppression pool temperature also reduce the wetwell airspace pressure, which would partially offset the benefit of the lower pool temperature.

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4.5 SUMMARY AND CONCLUSIONS

ATWS, SBO, and Appendix R events are highly unlikely events that are characterized by the failure of multiple design features. Failure assumptions in these events are beyond the original ECCS design basis; accordingly, additional failures such as loss of containment integrity, unless consistent with the credible consequential failure, are not imposed on the supporting design analyses.

Relaxation of typical conservatisms in analysis assumptions and input values to represent expected event responses should be considered.

The net positive suction head available determinations will be completed on plant specific bases. It is expected that the deterministic approach utilizing nominal input values will be used to calculate $NPSH_a$ for special events. Should this approach not satisfactorily show that $NPSH_a - NPSH_r \geq 0$; then the statistical approach utilizing the mean output values will be used to show the expected realistic response to the event.

5.0 OTHER EVALUATIONS ASSOCIATED WITH NPSH

5.1 EFFECTS OF REDUCED NPSH ON PUMP PERFORMANCE

When a pumped fluid enters the eye region of the pump impeller there is a localized pressure drop with respect to the pump suction and discharge pressures. When a liquid flows into a region where its pressure is reduced to its vapor pressure the liquid will begin to change state and vapor pockets will develop within the liquid. These vapor pockets (bubbles) are carried along within the liquid unit a region of higher pressure is reached, where they suddenly collapse. This phenomenon is called cavitation.

The Hydraulic Institute standard criterion for establishing the required net positive suction head value is where cavitation causes the total dynamic head to be reduced by 3%. Pump manufacturers can determine the required net positive suction pressure for an individual pump through testing and typically show the $NPSH_r$ data on the pump performance curves.

As $NPSH_a$ is lowered below the $NPSH_r$ the percentage of liquid that undergoes state change and becomes vapor increases. This will cause increased cavitation and further reduce the total dynamic head of the pump. The effects will be flow surging, and increased noise and vibration levels at the pump. As the $NPSH_a$ is further reduced at a specific pump flow rate, a condition called head collapse will be entered. This condition is where the percentage of liquid that is in vapor phase is so great that pump flow ceases.

A pump manufacturer, Sulzer Pumps, has conducted certified pump performance tests at reduced NPSH (Reference 10). Some of these tests ran pumps for extended periods (2-3 hours) at 1% to 6% head loss without losing suction; despite flow surging, noise and increased vibration experienced, the post-test inspections showed no damage to the inspected impellers. Several tests included $NPSH_a$ reduction to initiate loss of suction (head collapse). Pumps were shown to recover, without visible damage, after NPSH was restored.

A simple way to maintain $NPSH_a$ greater than $NPSH_r$ during an accident or special event when physical conditions like suppression pool temperature increases, suction strainer debris loading increases, or containment pressure decreases are the cause of the $NPSH_a$ reduction, is to throttle the discharge flow of the pump. This action will increase the $NPSH_a$ by reducing the pump flow rate or velocity to reduce friction in the suction line including the suction strainer and decrease the $NPSH_r$ by driving the pump operating point into a region of lower $NPSH_r$ (Reference 11). Pumps having aggressively designed suction characteristics (high suction specific speed) will have an $NPSH_r$ curve that increases as the flow decreases, with the lowest value being near the best efficiency point (BEP); in this case lowering the flow would cause the situation to worsen. Additionally, if non-catastrophic pump damage were to occur, reducing pump dynamic head and flow capabilities, many ECCS pumps would still be capable of meeting their required mission functions.

Minor excursions are not expected to lead to gross failures; even if pump damage does occur, ability to achieve design function is not reduced.

5.2 ALTERNATE METHODS TO CONTAINMENT OVERPRESSURE CREDIT

Information on this topic is supplied in Appendix C and Appendix D. In general, the alternate methods are not practical or are prohibitively costly to implement. Consequently, it is not anticipated that licensees will implement enhancements or alternatives to COP.

5.3 RISK ASSESSMENT

The risk analysis assesses the impact on plant risk if containment overpressure is assumed not present (e.g., postulated pre-existing primary containment failure) during the postulated accident scenarios such that adequate Low Pressure (LP) ECCS pump NPSH is available.

NRC Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 12), provides guidance for determining the risk impact of plant-specific changes to the licensing basis.

This risk assessment addresses Principle #4 of the RG 1.174 risk-informed structure. Principle #4 of RG 1.174 involves the performance of a risk assessment to show that the impact on the plant core damage frequency (CDF) and large early release frequency (LERF) risk metrics due to the proposed change are within acceptable ranges, as defined by RG 1.174.

The DBA-LOCA risk analyses in this assessment are sufficiently generic and conservative such that the results are applicable to the BWR fleet. Non-LOCA events are also considered in this analysis in a simplified fashion to bound the BWR fleet.

BWR plants should review and verify that this assessment bounds their plant.

5.3.1 Risk Assessment Approach

This risk assessment is performed by modification and quantification of the at-power internal events probabilistic risk assessment (PRA) models, and using the risk assessment guidance of NRC RG 1.174.

RG 1.174 states that the scope of the analysis shall be commensurate with the risk impact of the requested change. If the risk impact is very small, as is expected for COP, explicit analysis of external events is not required and the assessment of external events can be performed qualitatively. If the plant maintains external events PRA models (e.g., internal fire PRA, seismic PRA) then those models may also be manipulated to support the analysis. However, most U.S. BWRs currently do not maintain external events PRAs; as such, the COP risk assessment focuses on quantification of the at-power internal events PRA. External events and special events are addressed here as conservative sensitivity cases.

5.3.1.1 Use of MNGP PRA

The current MNGP 2005 PRA models are used as input to perform the example COP risk assessment. The MNGP PRA uses widely accepted PRA techniques for event tree and fault tree analysis. Event trees are constructed to identify core damage and radionuclide release sequences. The event tree "top events" represent systems (and operator actions) that can prevent or mitigate core damage. Fault trees are constructed for each system in order to identify the failure modes.

Analysis of component failure rates (including common cause failures) and human error rates is performed to develop the data needed to quantify the fault tree models.

5.3.1.2 PRA Quality

RG 1.174 requires a determination that the PRA is of sufficient quality (both in scope and level of detail) to support the analysis. Such a quality assessment is supported on a plant specific basis by discussion of any independent peer reviews of the PRA and/or by comparison to industry PRA Standards (e.g., America Society of Mechanical Engineers (ASME) PRA Standard – Reference 13).

The MNGP PRA used as input to this analysis is of sufficient quality and scope for this application. The MNGP PRA is highly detailed, including a wide variety of initiating events (e.g., transients, internal floods, LOCAs inside and outside containment, support system failure initiators), modeled systems, operator actions, and common cause events.

The MNGP at-power internal events PRA received a formal industry PRA Peer Review. All of the "A" and "B" priority comments from the Peer Review have been addressed.

RG 1.174 directs that the quality of a PRA analysis used to support an application "be commensurate with the application for which it is intended." The example plant PRA used in this analysis, combined with the conservative assumptions used in this assessment, are such that the quality of this PRA analysis is sufficient for this application to show that the risk impact of COP for low pressure ECCS NPSH is "very small" as defined by RG 1.174. The issue of COP is sufficiently narrow in scope with respect to accident sequence modeling, fault tree logic, and data such that the level of quality of an industry PRA with respect to RG 1.200 or other PRA standards needs to address only a narrow set of supporting requirements. A BWR plant PRA need not be fully compliant with all aspects of RG 1.200 (Reference 14) in order that the results of this analysis remain applicable. Those PRA elements key to this risk application are:

- Large LOCA initiating event frequency
- Large LOCA accident sequence structure
- Containment isolation fault tree

The quality of these elements in the example plant risk assessment in this study is sufficient for this risk application.

The Large LOCA initiating event frequency in the example plant PRA (1.64E-4) is at the high end of the frequency range compared to recent industry initiating event studies. NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995" (Reference 15), recommends a BWR Large LOCA initiating event frequency of 3E-5/yr. NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants" (Reference 16), recommends a BWR Large LOCA initiating event frequency of 7.0E-6/yr. Use of these lower Large LOCA frequency estimates would reduce further the already "very small" risk change calculated in this study.

The COP for DBA LOCAs impacts a single accident sequence in the example plant PRA. That sequence is: "Large LOCA Initiator" x "Vapor Suppression System Success" x "Low Pressure Injection Failure". The quality of this sequence structure is suitable for this analysis; the appropriate safety functions are modeled and no optimistic or controversial assumptions are made in this simple sequence. Only LPCI and Core Spray are credited as adequate core cooling options; no alternate RPV injection sources are credited. As such, this simple accident sequence structure in this example plant analysis bounds BWR industry PRAs.

The containment isolation fault tree for the example plant PRA is of sufficient quality for this analysis. The fault tree models the containment main penetration lines that communicate with the containment atmosphere, and includes both random and common cause failures of the containment isolations valves in each modeled penetration. The probability of a pre-existing containment leak was added to the example plant containment isolation fault tree. As discussed later, the pre-existing containment leakage probability used in this example plant analysis dominates the containment isolation functional failure probability. Therefore, the level of detail in the containment isolation fault tree regarding failure of penetrations to isolate is not significant to the results and conclusions of this study.

The human error probabilities used in this example plant analysis are sufficiently conservative that the quality of the human reliability analysis is not key to this analysis. As shown later, even assuming a 1.0 probability for failure to throttle the ECCS pumps does not change the conclusions that the risk impact of COP is "very small".

Steps to COP Risk Assessment

The following major analytical steps best describe the performance of the COP risk assessment:

- Assessment of NPSH calculations
- Estimation of pre-existing containment failure probability
- Analysis of relevant plant experience data
- Manipulation and quantification of PRA models
- Comparison to ΔCDF and ΔLERF RG 1.174 acceptance guidelines
- Performance of uncertainty and sensitivity analyses

These steps are discussed below.

5.3.2 Assessment of NPSH Calculations

The purpose of this task is to develop an understanding of the MNGP EPU NPSH calculations that result in the need to credit containment overpressure for Large LOCA accident scenarios.

The NPSH calculations are reviewed to understand the scenarios of interest that require COP to determine how best to modify the PRA models.

Two general approaches to PRA modeling of COP exist depending upon the number and types of NPSH calculations available:

1. Use of sensitivity studies of DBA NPSH calculations (not used for example plant)

2. Use of NPSH results from Monte Carlo process (used for example plant)

5.3.2.1 Use of Sensitivity NPSH Calculations

In the first case, sensitivity calculations would be performed of the deterministic DBA NPSH calculations to determine under what conditions of more realistic inputs is there no need for COP in the determination of low-pressure ECCS pumps NPSH. This approach is analogous to the use of multiple discrete thermal hydraulic calculations (e.g., using the MAAP code) in the PRA to define accident sequence and functional success criteria.

As indicated in the previous sections of this report, the following key plant parameters, among others, vary and can significantly impact available NPSH:

- Initial reactor power level
- Initial suppression pool temperature
- Initial suppression pool volume
- Ultimate heat sink temperature

By varying these and other, parameters and performing sensitivity calculations, the various conditions under which COP is not required can be determined.

It is recognized that there are potentially thousands of discrete combinations of more realistic calculation inputs that may show that COP is not necessary for maintenance of low pressure ECCS pump NPSH during DBA accidents. However, it is not practicable, nor necessary, to add such a large number discrete scenarios into the PRA logic model. The modeling of the varying states needs to be simplified to address the key variables, identify a reasonable number of representative scenarios, and to use exceedance probabilities for the various plant variables.

The plant specific NPSH sensitivity calculations will determine what combination and range of variables define when COP is required.

The representative scenarios would then be modeled into the accident sequence logic using standard fault tree modeling techniques to fail LP ECCS systems given a DBA-LOCA with an unisolated containment, and the probability that initial plant conditions (e.g., ultimate heat sink high temperature, SP initial high temperature) necessary to create inadequate NPSH exist at the start of the event. Analysis of plant specific operating experience and ultimate heat sink (UHS) temperature is used to determine the appropriate exceedance probabilities for the key variables (refer to Appendix A for the exceedance probabilities calculated for the example plant in this report).

This approach was not used in the example plant of this topical report; rather, the results of the Monte Carlo $NPSH_a$ analysis are used in the example analysis to define a single basic event to model the probability that plant conditions at the time of DBA large LOCA (LLOCA) result in inadequate LP ECCS NPSH (refer to discussion below).

5.3.2.2 Use of NPSH Result from Monte Carlo Analysis

As discussed in the previous sections, a Monte Carlo statistical analysis can be performed of the NPSH calculations to produce a 95/95 result for the available NPSH in a given accident scenario. The 95/95-point represents the 95% confidence level that the available NPSH is greater than the calculated Monte Carlo result with a 0.95 probability (or, that there is only a 0.05 probability that it is lower).

The Monte Carlo NPSH results can be used to define a single PRA basic event with a probability based on the Monte Carlo result. The basic event represents the probability that initial plant conditions (i.e., high initial suppression pool temperature, high UHS temperature, etc.) exist at the onset of the modeled DBA scenarios such that inadequate LP ECCS NPSH is available.

For the purpose of modeling the conditions of inadequate NPSH, the PRA is interested in the probability of the plant conditions such that $NPSH_a$ is less than $NPSH_r$ (i.e., the fraction of the NPSH spectrum below the $NPSH_a = NPSH_r$ point). The 95/95-point for $NPSH_a$ from the Monte Carlo analysis is not directly usable in the PRA logic modeling (i.e., use it directly as a 0.05 probability basic event) unless it coincidentally equals $NPSH_r$.

As the result of the Monte Carlo analysis is a single 95/95 NPSH point rather than a continuous probability distribution as a function of NPSH, engineering judgment is used (based on review of the NPSH Monte Carlo results) to assign an appropriate basic event probability for each DBA scenario that initial plant conditions exist at the onset of the scenarios. The results are as follows for the example plant (see Appendix A for a discussion of these scenarios):

- Scenario #1: 1.0E-1
- Scenario #2: 5.0E-1
- Scenario #3: 1.0E-1

For Scenario #1, the probability that plant conditions will result in inadequate NPSH is known to be some value higher than 5E-2 (i.e., because the 95/95 $NPSH_a$ point is below $NPSH_r$). As the calculated $NPSH_a$ 95/95 point is comparatively (1-2 ft.) close to $NPSH_r$ in the short time frame modeled, a nominal probability of 1E-1 is used. The same results apply to Scenario #3 (i.e., the calculated $NPSH_a$ 95/95 point is comparatively close to $NPSH_r$).

For Scenario #2, the calculated $NPSH_a$ 95/95 point is much lower (by a factor of 2-3) than it is for Scenarios #1 and #3. As such, a nominal probability of 5E-1 is used for Scenario #2.

As shown later with a quantitative sensitivity case, the exact values of the above probabilities are not necessary in showing that the COP risk impact is "very small".

The three scenarios for the example plant are summarized in Table 5-1. As can be seen from Table 5-1, Scenarios #1 and #3 may be modeled together as a single scenario because the impact of LPCI Loop Select Logic failure does not change the conclusion that COP is required in approximately 7 minutes and that throttling LP ECCS will preclude the need for COP. Therefore, the scenario modeling in the PRA for the DBA-LOCA is as follows for the example plant:

- <u>Scenario #1 / #3</u>: (Large LOCA Initiator) x (Suppression Pool Cooling (SPC) Not Initiated Within t=10 min.) x (Containment Isolation fails at t=0) x (Operators Fail to Throttle LP ECCS Flow Within t=10 min.) x (Probability that Existing Plant Conditions Result in Inadequate NPSH) x (Probability that pump failed due to inadequate NPSH).
- <u>Scenario #2</u>: (Large LOCA Initiator) x (One Division ECCS Available) x (SPC Not Initiated Within t=10 min.) x (Containment Isolation fails at t=0) x (Probability that Existing Plant Conditions Result in Inadequate NPSH) x (Probability that pump failed due to inadequate NPSH).

The probability that the LP ECCS pumps are failed due to inadequate NPSH is conservatively assumed to be 1.0 in the example plant analysis. For a specific plant analysis, pump survival considerations can be made.

The modeling of these scenarios in the example plant PRA is discussed in Section 5.3.5

5.3.3 Estimation of Pre-Existing Containment Failure Probability

This task involves defining the size of a pre-existing containment failure pathway to be used in the analysis to defeat the COP, and then quantifying the probability of occurrence of the un-isolable pre-existing containment failure. The approach to this input parameter calculation will follow EPRI guidelines regarding calculation of pre-existing containment leakage probabilities in support of integrated leak rate test (ILRT) frequency extension Licensing Amendment Requests (LARs) (i.e., EPRI Report 1009325, Risk Impact of Extended Integrated Leak Rate Testing Intervals - Reference 6). This is the same approach used in the Vermont Yankee EPU COP analyses presented to the ACRS in November and December 2005 (Reference 17).

Containment failures that may be postulated to defeat the containment overpressure credit include containment isolation system failures and pre-existing unisolable containment leakage pathways. The pre-existing containment failure may be one that only manifests itself as the containment pressurizes.

The containment isolation failures should already be modeled in the containment isolation fault tree used in the plant Level 2 PRA. These failures include failures on demand and failures of valves to remain closed during the standard 24-hour PRA mission time. If the containment isolation fault tree does not already contain a basic event for pre-existing containment leakage, such an event must be added to the fault tree logic as it is a dominant contributor to the probability of an unisolated containment..

The pre-existing containment leakage probability may be obtained from EPRI 1009325 (Reference 6). EPRI 1009325 provides a framework for assessing the risk impact for extending ILRT surveillance intervals. EPRI 1009325 includes a compilation of industry containment leakage events, from which an assessment was performed of the likelihood of a pre-existing unisolable containment leakage pathway.

A total of seventy-one (71) containment leakage or degraded liner events were compiled. Approximately half (32 of the 71 events) had identified leakage rates of less than or equal to 1 La (i.e., the Technical Specification containment allowed leakage rate). None of the 71 events had identified leakage rates greater than 21 La. EPRI 1009325 employed industry experts to review and categorize the industry events, and then various statistical methods were used to assess the data. The resulting probabilities as a function of pre-existing leakage size are summarized here in Table A-4.

The EPRI 1009325 study uses 100 La as a conservative estimate of the leakage size that would represent a large early release pathway consistent with the LERF risk measure, but estimated that leakages of 600 La or greater are a more realistic representation of a large early release. The COP risk assessment for the Vermont Yankee Mark I BWR plant, presented to the ACRS (Reference 17), determined a leakage size of 27 La using the conservative 10CFR50, Appendix K containment analysis approach. Earlier ILRT industry guidance (Nuclear Energy Institute

(NEI) Interim Guidance) conservatively recommended use of 10 La to represent "small" containment leakages and 35 La to represent "large" containment leakages.

This analysis is not concerned per se about the size of a leakage pathway that would represent a LERF release, but rather a leakage size that would defeat the containment overpressure credit. Given the low likelihood of such a leakage, the exact size is not key to this risk assessment, and no detailed calculation of the exact hole size is performed here. A sensitivity study discussed later assesses the sensitivity of the results to the pre-existing leakage size assumption.

Given the above, the base analysis here assumes 20 La as the size of a pre-existing containment leakage pathway sufficient to defeat the containment overpressure credit. Such a hole size does not realistically represent a LERF release (based on EPRI 1009325) and is also believed (based on the Vermont Yankee (VY) hole size estimate) to be on the low end of a hole size that would preclude containment overpressure credit. As can be seen from Table A-4, the probability of a 20 La pre-existing containment leakage at any given time at power is 1.88E-03.

This low likelihood of a significant pre-existing containment leakage path is consistent with MNGP primary containment performance experience. The MNGP primary containment performance experience shows MNGP containment leakages much less than 1 La.

Sensitivity studies to the base case quantification assess the sensitivity of the results to the pre-existing leakage size assumption.

5.3.4 Analysis of Relevant Plant Experience Data

An unisolated primary containment is not the only determining factor in defeating low-pressure ECCS pump NPSH. Variations in UHS and suppression pool water temperatures, suppression pool level and RHR heat exchanger "K" value at the Monticello plant were statistically analyzed. The purpose of this data assessment is to estimate realistic probabilities that UHS water temperature, suppression pool level and temperature, and heat exchanger effectiveness will exceed a given value, i.e. the probability of exceedance. These values are used as input into the Monte Carlo simulations of the available NPSH and in the risk assessment.

This step involves obtaining the plant experience data and performing statistical analysis to determine the probabilities of exceedance. Refer to Appendix A.

5.3.5 Manipulation and Quantification of PRA Models

This task is to make the necessary modifications to the PRA models to simulate the loss of low-pressure ECCS pumps during a Large LOCA. Large LOCA initiated sequences in the PRA are modified as appropriate to mirror the DBA accident calculations requiring COP. Accident sequences involving Interfacing Systems LOCAs and other LOCAs Outside Containment are not adjusted in this risk assessment because such LOCAs result in deposition of decay heat directly outside the containment and not into the suppression pool.

The modeling and quantification is performed consistent with common CAFTA modeling techniques for the example plant. For the example plant, the probability of pre-existing leakage was added to the Level 2 PRA containment isolation fault tree because it did not exist in the base PRA model.

PRA Model Modifications

The modifications made to the example plant PRA to model the COP for DBA LOCA scenarios are shown in Figure 5-1 and Figure 5-2 show the DBA LOCA COP scenario logic developed under a sub-tree that is input into the CS and LPCI fault tree logic. Figure 5-3 shows the pre-existing containment leakage basic event added to the containment isolation fault tree.

This figure shows that pumps were conservatively assumed to fail after 10 minutes, though realistically the pumps would be expected to last many hours in cavitation where $NPSH_a$ is less than $NPSH_r$.

As can be seen in Figure 5-1, the new logic utilizes an AND gate with the large LOCA initiator to ensure that the logic applies to large LOCA initiated accident sequences. If the plant has multiple large LOCA inside containment initiators (i.e., it is not uncommon for BWR PRAs to have multiple Large LOCA initiators that model specific break locations), then the various large LOCA initiators would be inserted under an OR gate in place of the large LOCA initiator basic event shown in Figure 5-1.

Initiation of SPC is a manual action. Therefore, the basic event stating that SPC is not initiated within t=10 minutes is assigned a 1.0 probability. The time to align SPC typically takes 5-10 minutes and would be done under high stress during a DBA-LOCA.

The two basic events that model the probability that plant conditions at the time of the DBA LOCA contribute to inadequate LP ECCS are based on the discussions in Section 5.3.2.

The human error probability basic event for operator failure to throttle LP ECCS is calculated using the same human reliability analysis methodology (i.e., NUREG/CR-4772 – Reference 18) used in the example plant PRA:

- Per the plant Emergency Operating Procedures (EOPs) and operator training, the operators will throttle ECCS flow as necessary per NPSH curves existing on the Emergency Operating Procedure (EOP) flowcharts
- The time of the initial cue to the operators for the need to throttle ECCS flow is estimated at t=5 minutes for Scenarios 1 & 3. This is the point at which available head is nearing $NPSH_r$ and which flow fluctuations may be notable to the operator.
- The end of the available time window to the operator is conservatively estimated at t=10 minutes. In addition, this is the time at which pump head collapse is assumed to occur. This time is judged conservative for the example plant.
- Manipulating LP ECCS pump flow is a manual action performed at the main control panels in the control room. The time required to travel to the proper panel(s) and perform the flow manipulation is estimated at 1 min.
- Therefore, the available diagnosis time to the operator is (10 min. – 5 min.) – 1 min. = 4 min.
- Using the example plant Human Reliability Analysis (HRA) Methodology (Reference 18), the diagnosis error contribution for a diagnosis time frame of 4 min. is 2.5E-1; and the manipulation error rate for performing the action is 5E-3. The total Human Error Probability (HEP) for failure to throttle is 2.55E-1.

In conditions of inadequate NPSH, the pumps will experience surging and cavitation but will not necessarily fail. However, the example plant analysis conservatively assumes the low-pressure ECCS pumps fail with a probability of 1.0 given inadequate NPSH and failure to throttle.

The probability of an unisolated containment at the time of the accident is modeled using the existing containment isolation fault tree present in the example PRA. A basic event for the probability of a pre-existing containment leak at the time accident was added to the containment isolation fault tree. The probability of the pre-existing leakage basic is discussed previously in Section 5.3.3 and is based on an assumed hole size of 20 La.

Scenario #2 involves failures that result in only one available ECCS division. Those failures are a loss of offsite power (LOOP) combined with failure of one division of ECCS (the DBA single failure is assumed to be an emergency diesel generator (EDG), but the PRA recognizes that it could also be a bus or ECCS equipment failures):

- The conditional probability of a LOOP given a LOCA initiator is 2.4E-2, based on Reference 19, USNRC Memorandum to Samuel J. Collins, Director Office of Nuclear Reactor Regulation, from Ashok C. Thadani, Director of Nuclear Regulatory Research, "Transmittal of Technical Work to Support Possible Rulemaking on a Risk-Informed Alternative to 10 CFR 50.46/GDC 35", July 31, 2002.
- Failure of one division of ECCS is modeled as failure of Division 1 "OR" Division 2 ECCS. Each division is modeled with an undeveloped basic event with a probability of 1E-1. This 1E-1 probability covers failure of one EDG (a contribution of approximately 5E-2), failure of the associated safety bus (a negligible contribution), and failures for one division of ECCS pumps and valves (a contribution of approximately 5E-3), and is judged conservative.

<u>PRA Model Quantification</u>

The Level 1 (core damage) PRA is then quantified using the standard quantification techniques of the base PRA. The impact on the Level 2 LERF accident sequences should be modeled with the assumption that the COP failure scenarios lead directly to a LERF release. This approach is consistent with the modeling and NPSH calculation assumptions of loss of containment integrity at t=0. As such, the ΔLERF should be assumed to equal the calculated ΔCDF. It is not necessary to perform an explicit LERF model quantification using this assumption.

The size of the assumed containment hole used in the pre-existing containment leakage basic event is conservatively small (i.e., BWR PRAs typically use a 2" diameter hole in the primary containment to represent the minimum size of a LERF release pathway, and a 2" diameter hole is much greater than the 20 La equivalent hole size used in the base calculation). In addition, the location of the assumed containment leakage pathway has an impact on LERF. If the containment leakage pathway is assumed to exist in the wetwell airspace then the post-accident releases from the containment would be scrubbed by the suppression pool and thus not result in a LERF magnitude release. This example plant analysis conservatively assumes that the containment leakage pathway is such that, given a core damage event, the conditional probability of a LERF release is 1.0. The impact of this conservative assumption on ΔLERF does not change the overall conclusion that the risk impact of COP is very small.

5.3.6 Comparison to ΔCDF and ΔLERF RG 1.174 Acceptance Guidelines

The revised MNGP PRA models are quantified to determine the change in the base CDF. As discussed above in Section 5.3.5, the change in LERF should be assumed to equal the change in CDF.

The RG 1.174 ΔCDF and ΔLERF risk acceptance guidelines are summarized in Figures 5-4 and 5-5, respectively. The boundaries between regions are not necessarily interpreted by the NRC as definitive lines that determine the acceptance or non-acceptance of proposed license amendment requests; however, increasing delta risk is associated with increasing regulatory scrutiny and expectations of compensatory actions and other related risk mitigation strategies.

The risk impact results for the example BWR plant for COP for DBA-LOCAs is:

- ΔCDF = 9.0E-9
- ΔLERF = 9.0E-9

Both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region.

These impacts are referenced with respect to the base modeling assumption that no COP is required for LP ECCS adequate NPSH during DBA-LOCA scenarios. If the base model where revised to include modeling of any existing COP already allowed at the plant, the change in risk for the additional COP required by an EPU (or other Licensing Amendment Request) would be even smaller.

5.3.7 Uncertainty and Sensitivity Analyses

To provide additional information for the decision making process, the risk assessment should be supplemented by parametric uncertainty analysis and quantitative and qualitative sensitivity studies to assess the sensitivity of the calculated risk results.

Uncertainty is typically categorized into the following three types, consistent with PRA industry literature:

- Parametric
- Modeling
- Completeness

Parametric uncertainties are those related to the values of the fundamental parameters of the PRA model, such as equipment failure rates, initiating event frequencies, and human error probabilities. Typical of standard industry practices, the parametric uncertainty aspect is assessed by performing a Monte Carlo parametric uncertainty propagation analysis. Probability distributions are assigned to each parameter value in the PRA, and a Monte Carlo sampling code is used to sample each parameter and propagate the parametric distributions through to the final results.

Modeling uncertainty is focused on the structure and assumptions inherent in the risk model. The structure of mathematical models used to represent scenarios and phenomena of interest is a source of uncertainty, because models are a simplified representation of a real-world system.

Model uncertainty is addressed here by the identification and quantification of focused sensitivity studies.

Completeness uncertainty is primarily concerned with scope limitations. Scope limitations are addressed here by the qualitative assessment of the impact on the conclusions if external events and shutdown risk contributors are also considered.

5.3.7.1 Parametric Uncertainty Analysis

The MNGP PRA is not currently constructed to allow parametric uncertainty analysis; as such, parametric uncertainty analysis was not performed. However, based on knowledge of the issues, the COP risk impact, and PRA parametric uncertainty assessments, the results of a parametric uncertainty analysis would not change the conclusion that the risk impact of COP for DBA-LOCAs is "very small" per RG 1.174.

5.3.7.2 Modeling Uncertainty Analysis

As stated previously, modeling uncertainty is concerned with the sensitivity of the results due to uncertainties in the structure and assumptions in the logic model. EPRI has developed a guideline for modeling uncertainty that takes the rational approach of identifying key sources of modeling uncertainty and then performing appropriate sensitivity calculations. This approach is taken here.

The modeling issues selected here for assessment are those related to the risk assessment of the containment overpressure credit. This assessment does not involve investigating modeling uncertainty with regard to the overall base PRA. The modeling issues identified for sensitivity analysis are:

- Pre-existing containment leakage size and associated probability
- Calculation of containment isolation system failure
- Probability of plant conditions contributing to inadequate NPSH
- Large LOCA initiator
- Throttling HEP

Sensitivity Case 1: Pre-Existing Containment Leakage Size/Probability

The base case analysis assumes a pre-existing containment leakage pathway leakage size of 20 La that would result in defeat of the necessary containment overpressure credit.

A larger pre-existing leak size of 100 La, consistent with the EPRI 1009325 recommended assumption for a "large" leak, is used in this sensitivity to defeat the necessary COP. From EPRI 1009325, the probability of a pre-existing 100 La containment leakage pathway at any given time at power is 2.47E-04. As such, the base case value of 1.88E-3 (applicable to 20 La) for the probability of a pre-existing containment leakage sufficient to defeat the necessary COP is changed to 2.47E-4 (applicable to 100 La) in this sensitivity case.

Sensitivity Case 2: Calculation of Containment Isolation System Failure

The base case quantification uses the containment isolation system failure fault tree logic to represent failure of the containment isolation system. The fault tree specifically analyzes primary containment penetrations greater than 2" diameter. This modeling sensitivity case expands the scope of the containment isolation fault tree to include smaller lines as potential defeats of COP. This sensitivity is quantified by multiplying by a factor of 10 the probability contribution in the containment isolation fault tree from isolation of penetrations in response to a containment isolation signal.

Sensitivity Case 3: Probability of Plant Conditions Contributing to Inadequate NPSH

The basic event probabilities for the different scenarios that plant conditions at the time of the DBA-LOCA contribute to inadequate LP ECCS are based on the discussions in Section 5.3.2. As previously discussed, precise estimates of these probabilities are not necessary to show that the risk impact of COP for LP ECCS NPSH is very small. This fact is shown by this sensitivity. This sensitivity is performed assuming that plant conditions (e.g., high initial suppression pool temperature, high UHS temperature, etc.) contributing to inadequate NPSH exist 100% of the time.

Sensitivity Case 4: Large LOCA Initiators in the PRA

The example plant has a single "Large LOCA" initiator in the PRA, and this initiator was used to represent the DBA LOCA scenarios. However, in addition to the "Large LOCA" initiator, the example plant PRA also contains an initiator for "RPV Rupture" and LOCA-induced scenarios caused by Transient initiators with failure of SRVs to actuate. This sensitivity case includes the "RPV Rupture" initiator and the LOCA-induced scenarios (i.e., isolation transients in which all SRVs fail to open) in the COP risk assessment. The impact on the base results is negligible.

Sensitivity Case 5: LP ECCS Throttling HEP

The example plant analysis uses a human error probability, HEP, of 2.55E-1 for failure to throttle LP ECCS to avoid pump failure due to inadequate NPSH. This HEP is based on the plant specific timings from the thermal hydraulic calculations and the human reliability analysis methodology used in the example plant PRA. This sensitivity study conservatively assumes that the HEP for failure to throttle LP ECCS is 1.0.

Summary of Modeling Uncertainty Results

The results of these sensitivity studies are as follows:

Case	ΔCDF	ΔLERF
Sensitivity Case 1	1.2E-9	1.2E-9
Sensitivity Case 2	1.4E-8	1.4E-8
Sensitivity Case 3	8.4E-8	8.4E-8
Sensitivity Case 4	9.0E-9	9.0E-9
Sensitivity Case 5	3.3E-8	3.3E-8

The above sensitivity studies do not change the base conclusions that the risk impact of COP for a DBA-LOCA is "very small" per RG 1.174.

5.3.7.3 Completeness Uncertainty Analysis

As stated previously, completeness uncertainty is addressed here by the qualitative assessment of the impact on the conclusions if special events, external events and shutdown risk contributors are also considered.

ATWS

COP is also required at some BWRs for ATWS scenarios.

The risk impact of COP for low pressure ECCS pump NPSH during ATWS scenarios can be assessed with the following representative ATWS scenario:

- Initiator: Isolation event
- Failure to SCRAM
- Successful RPV level/power control
- Containment isolation failure at t=0
- Only one division of ECCS available
- Operators fail to throttle ECCS pumps

An isolation event is one that results in isolation of the RPV from the main condenser heat sink. Based on NUREG/CR-6928 (Reference 16), the industry average frequency for such an event is approximately 2E-1/yr. Based on the various isolation initiating events (e.g., Main Steam Isolation Valve Closure (MSIVC), Loss of Condenser Vacuum, etc.) modeled in the example plant PRA, the frequency of such an event at MNGP is approximately 3E-1/yr. The frequency of 3E-1/yr is used in this analysis.

The probability of scram failure in the example plant PRA is 5.9E-6. This probability is consistent with other current BWR industry PRAs.

The sum of Alternate Rod Insertion (ARI), Recirculation Pump Trip (RPT), Standby Liquid Control (SLC), and operator level control and Automatic Depressurization System (ADS) inhibit action failures is generally in the range of 0.1 to 0.2 for industry BWR PRAs, which would result in a probability of successful level/power control in the 0.8 to 0.9 range. This analysis conservatively assumes the probability of successful level/power control is 1.0. Failure of level/power control would result in a scenario, which would lead to core damage regardless of COP issues; therefore, such scenarios are not part of this assessment.

The probability of containment isolation failure at t=0 is approximately 2E-3 (based on the example plant PRA).

As discussed earlier in the base case analysis of this risk assessment, the failure probability for one division of ECCS is approximately 5E-3.

The same human error probability of 2.55E-1 used in the example plant analysis for failure to throttle the ECCS pumps is assumed here.

The risk impact for such a scenario is calculated as:

$$3E\text{-}1 \times 5.9E\text{-}6 \times 1.0 \times 2E\text{-}3 \times 5E\text{-}3 \times 2.55E\text{-}1 = 4.5E\text{-}12/yr$$

Even if this representative ATWS scenario were to require only that the containment be unisolated (i.e., failure of one division of ECCS not assumed and throttling not a success path), the accident sequence frequency would still be a non-significant 3.5E-9/yr.

Postulating this additional scenario would not change the conclusion that the risk impact of COP is "very small" per RG 1.174.

SBO

COP is also required at some BWRs for SBO scenarios.

The risk impact of COP for low pressure ECCS pump NPSH during SBO scenarios can be assessed with the following representative SBO scenario:

- Initiator: Loss of Offsite Power
- Failure of all EDGs
- One SBO capable injection source successfully operates
- Containment isolation failure at t = 0
- Offsite alternating current (AC) power recovered at t = 4 hours (the example plant SBO coping period)
- Alignment of SPC at t = 4hrs

Based on NUREG/CR-6928 (Reference 16), the industry average frequency for loss of offsite power is approximately 4E-2/yr. The LOOP initiator frequency in example plant PRA is 2.28E-2/yr. The frequency of 4E-2/yr is used in this analysis.

As discussed previously for the base case analysis, the failure probability of one EDG is approximately 5E-2. Failure of all EDGs is estimated here using a common cause failure approach and assuming a 5% failure of all EDGs given failure of one EDG. The 5% common cause failure probability is conservative (industry average is in the 2-3% range). Therefore, the failure of all EDGs is estimated at 5E-2 x 0.05 = 2.5E-3.

This analysis assumes that the probability of a SBO capable injection source (e.g., Reactor Core Isolation Cooling (RCIC)) successfully operating for the SBO coping period is 1.0.

The probability of containment isolation failure at t=0 is approximately 2E-3 (based on the example plant PRA).

Based on NUREG/CR-6890 (Reference 20), Reevaluation of Station Blackout Risk at Nuclear Power Plants (Reference 20), the industry average exceedance probability for successfully recovering offsite AC at 4 hours following a LOOP at power is approximately 0.84. Failure of AC power recovery would result in a scenario, which would lead to core damage regardless of COP issues; therefore, such scenarios are not part of this assessment.

This analysis assumes that the probability of alignment of ECCS pumps to the suppression pool immediately following offsite AC recovery is 1.0. This assessment also assumes that throttling of the pumps will not prevent the inadequate NPSH condition and that the pumps fail with a probability of 1.0 once they are aligned to the pool.

This SBO scenario conservatively does not credit other injection systems (e.g., RCIC from the Condensate Storage Tank (CST); alternative RPV injection sources) that would be available after 4 hours when offsite AC power is recovered.

The risk impact for such a scenario is calculated as:

$$4E\text{-}2 \times 2.5E\text{-}3 \times 1.0 \times 2E\text{-}3 \times 0.84 \times 1.0 = 1.7E\text{-}7/yr$$

Postulating this additional scenario would not change the conclusion that the risk impact of COP is "very small" per RG 1.174 for the Core Damage Frequency risk metric. The Large Early Release Frequency risk metric is just above the border of the "very small" and "small" region. Relaxation of the excess conservatisms in the LERF modeling (e.g., recognizing that loss of low pressure ECCS at t = 4 hrs does not directly result in a LERF release) would show that LERF risk metric is also clearly in the "very small" region of RG 1.174.

<u>Seismic</u>

The change in plant risk due to seismic-induced large LOCA COP scenarios is non-significant and likely undetectable with current state of the technology seismic PRA. The COP scenarios require one or more RHR pumps to be in operation (i.e., the PRA already models core damage accident sequences in which loss of all RHR pumps causes loss of LP ECCS – due to the need to initiate emergency containment venting) and the containment to fail.

A seismic event severe enough to fail the primary containment will also fail, with a much higher likelihood, the RHR system. Another aspect is in the modeling of like component failures in a seismic PRA. In a seismic PRA, like components located on the same elevation (e.g., RHR pumps) are modeled as all failed given one fails. As such, if a seismic event fails an RHR pump (with some probability that varies depending upon the seismic magnitude), a seismic PRA will fail all the RHR pumps. As such, the likelihood of a seismic scenario that fails the containment yet fails only 2 or 3 out of the four RHR pumps is a very low likelihood scenario. As a final point on this issue, very high magnitude earthquakes become moot for this issue, as they would result in failure of key buildings and structures and lead directly to core damage.

As such, seismic issues do not impact the decision making for containment overpressure credit.

<u>Internal Fires</u>

COP for the DBA LOCA scenario is necessary, among other aspects, due to the large heat addition to the suppression pool during the blowdown. An internal fire-induced large LOCA type scenario (i.e., a scenario with large heat addition to the suppression pool and no high pressure injection sources available) can be postulated as follows:

- Initiator: Fire in main control panel initiates ADS, "OR" fire-induced isolation event with subsequent multiple stuck open relief valves
- Containment isolation failure at t=0
- Plant conditions at time of event contribute to inadequate NPSH
- Operators fail to throttle ECCS pumps

The fire-induced initiator can be estimated at 1E-4/yr. A fire in the main control panel that initiates ADS would be estimated in the 1E-5/yr to 1E-4/yr range using current industry fire initiator techniques. A fire induced isolation transient with subsequent multiple stuck-open relief valves (SORVs) is in the 1E-6/yr to 1E-4/yr range (i.e., the sum of all fire-induced isolation transients would be in the 1E-2/yr to 1E-4/yr range, and the probability of multiple SORVs given an isolation transient is approximately 1E-3). Therefore, the sum of both these two fire scenarios is estimated at 1E-4/yr.

The probability of containment isolation failure at t=0 is approximately 2E-3 (based on the example plant PRA). The example plant analysis does not assume that this fire scenario also results in fire-induced containment isolation failure. A fire in the control room that causes both a fire-induced ADS actuation and fire-induced containment isolation failure would involve fires initiating in separate control panels at the same time (an extremely low likelihood scenario). A postulated fire scenario in which a fire initiates in one panel and then the operators fail to suppress the fire such that it spreads to multiple panels would be modeled in a fire PRA as a control room evacuation scenario and would lead to core damage with a high conditional probability regardless of COP impacts.

The same probability of 1E-1 used for Scenario #1 for plant conditions at the time of the event that contribute to inadequate NPSH can be reasonably used here.

Likewise, the same human error probability of 2.55E-1 used in the example plant analysis for failure to throttle the ECCS pumps can also be assumed here. Use of this HEP assumes that the timing for the need for COP in this scenario occurs as fast for this fire-induced SORV event as it does for the DBA LOCA.

The risk impact for such a scenario is calculated as: 1E-4 x 2E-3 x 1E-1 x 2.55E-1 = 5.1E-9/yr

Although not a DBA LOCA, postulating this additional scenario would not change the conclusion that the risk impact of COP for a DBA LOCA is "very small" per RG 1.174.

Other External Hazards

In addition to seismic events and internal fires, the other following external hazard categories exist:

- High Winds/Tornadoes
- External Floods
- Transportation and Nearby Facility Accidents
- Other External Hazards

The NRC Individual Plant Examination of External Events (IPEEE) Program has generally determined that these other external hazard categories are not significant risk contributors. As such, these other external hazards are judged not to significantly impact the decision making for containment overpressure credit.

Shutdown Risk

The credit for containment overpressure is not required for accident sequences occurring during shutdown. As such, shutdown risk does not influence the decision making for containment overpressure credit.

5.3.8 COP Risk Assessment Conclusions

The risk impact results for the example BWR plant for COP for LP ECCS NPSH for DBA LOCAs is:

- ΔCDF = 9.0E-9
- ΔLERF = 9.0E-9

Both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region. These impacts are referenced with respect to the base modeling assumption that no COP is required for LP ECCS adequate NPSH during DBA LOCA scenarios. If the base model where revised to include modeling of any existing COP already allowed at the plant, the change in risk for the additional credit required by an EPU (or other Licensing Amendment Request) would be even smaller.

Sensitivity studies show that even assuming plant conditions (e.g., high suppression pool temperature, high UHS temperature, etc.) contributing to inadequate NPSH exist 100% of the time results in a "very small" calculated risk impact.

The results for COP for DBA LOCA scenarios are orders of magnitude below the upper threshold of the RG 1.174 "very small" risk increase region. Even if COP were assumed required for DBA LOCAs, special events, and external events, the conservative and simplified calculations in this analysis shows that the overall impact (i.e., summing the impacts of COP for all such accidents) would still remain within the 'very small" risk increase region of RG 1.174 for Core Damage Frequency and just above the border of the "very small" and "small" region for Large Early Release Frequency. Relaxation of the excess conservatisms in the LERF modeling in this analysis (e.g., recognizing that loss of low pressure ECCS at t=4 hours does not result in a LERF release) would show that LERF risk metric is also clearly in the "very small" region even when COP impacts for DBA LOCAs, special events, and external events are summed.

<u>Applicability to BWR Fleet</u>

The conclusion of this risk assessment (i.e., the risk impact of COP for DBA LOCA LP ECCS NPSH calculations is "very small" per RG 1.174 criteria) is applicable to the BWR fleet. The applicability of this conclusion to the BWR fleet is due to the following considerations:

- Large LOCA initiator frequencies in industry BWR PRAs are within the same order of magnitude as the example plant PRA used in this analysis.
- Containment isolation failure (which defeats COP) is dominated in the example plant PRA by the conservatively small pre-existing containment leakage assumed. The probability of pre-existing containment leakage is based on generic BWR industry data. This conservatively sized leakage probability will dominate the containment isolation failure probability in other BWR PRAs as well (i.e., the contribution of random and common cause isolation failures of penetrations lines is approximately an order of magnitude lower than the pre-existing containment leakage probability assumed in this risk assessment).
- The HEP for failure to throttle LP ECCS is based on plant specific timings from the thermal hydraulic calculations and the human reliability analysis methodology used in the example plant PRA. A sensitivity study performed for the example plant shows

that even assuming an HEP of 1.0 (i.e., 100% failure to throttle pumps) does not change the conclusion that the risk impact is "very small"

- Plant-specific experience with suppression pool temperature, UHS temperature, etc. will influence the estimated probability that existing plant conditions at the time of the DBA LOCA are sufficient to result in inadequate LP ECCS NPSH. A sensitivity study performed for the example plant shows that even assuming such conditions exist 100% of the time does not change the conclusion that the risk impact is "very small".

BWR plants should review and verify that this assessment bounds their plant.

5.4 DEFENSE IN DEPTH

The BWROG assessed the impact of crediting containment overpressure on defense-in-depth using NUREG-0800, Standard Review Plan 19.2 Section III.2.1.1.1 (Reference 29), which requires consideration of four objectives:

Objective #1: The change does not result in a significant increase in the existing challenges to the integrity of barriers.

This objective is met. Crediting containment overpressure does not introduce new initiators.

Objective #2: The proposal does not significantly change the failure probability of any individual barrier.

This objective is met in the following three ways:

- Previous example indicates very small ΔCDF, so there is insignificant change in the failure probability of the first barrier
- No impact on the reactor coolant system integrity, so there is no change in the failure probability of the second barrier
- No impact on containment integrity, so there is no change in the failure probability of the third barrier

Objective #3: The proposal does not introduce new or additional failure dependencies among barriers that significantly increase the likelihood of failure compared to the existing conditions.

This objective is met. Crediting containment overpressure does introduce dependency between the first barrier (fuel clad) and the third barrier (containment). However, previous examples indicate very small ΔCDF, so there is an insignificant increase in the likelihood of failure as compared to existing conditions.

Objective #4: The overall redundancy and diversity among the barriers is sufficient to ensure compatibility with the risk acceptance guidelines.

This objective is met. The previous example indicates that there is a very small ΔCDF as per the RG 1.174 risk acceptance guidelines.

The scoping risk evaluation did not identify any special circumstances that rebut the presumption of adequate protection provided by meeting the deterministic requirements and regulations.

5.5 SUMMARY AND CONCLUSIONS

The effects of reduced $NPSH_a$ below the $NPSH_r$ will cause increased cavitation and reduction in the total dynamic head of the pump. The effects will be flow surging, increased noise and vibration levels at the pump. As the net positive suction head available is further reduced, a condition call head collapse will be entered. This condition is where the percentage of liquid that is in vapor phase is so great that pump flow ceases.

Pump tests have been performed for extended periods where the $NPSH_a$ was substantially, below $NPSH_r$ and in some cases led to head collapse. Pumps were shown to recover, without visible damage, after $NPSH_a$ was restored.

The risk impact results for the example BWR plant for COP for DBA-LOCAs is ΔCDF = 9.0E-9, and ΔLERF = 9.0E-9.

Both the change in CDF and the change in LERF fall within the RG 1.174 "very small" risk increase region. Conservative evaluations for internal fires, Station Blackout and ATWS have changes in CDF and changes in LERF that fall within the RG 1.174 "very small" risk increase region. Seismic and other external hazards were categorized as non-risk significant contributors for COP.

The conclusion of this risk assessment (i.e., the risk impact of COP for ECCS NPSH calculations is "very small" per RG 1.174 criteria) is applicable to the BWR fleet based on the following considerations:

- Large LOCA initiator frequencies in industry BWR PRAs are within the same order of magnitude as the example plant PRA used in this analysis.
- Containment isolation failure (which defeats containment overpressure) is dominated by the pre-existing containment leakage, which is based on generic BWR industry data.
- Use of industry data and conservative failure probabilities.
- Sensitivity study performed to bound plant contributing conditions and human error probability.

Plant submittals should determine and document that their plant risk assessments are bounded by the generic analysis, or provide plant-specific risk analysis.

Table 5-1
Summary of Understanding of DBA - LOCA NPSH Issues for Use in PRA Modeling
(Assuming no COP Exists)

			@ t = 0 min.			@ t = 10 min.					
Scenario	**Initiating Event (IE)**	**Single Failure**	**LP ECCS Pumps Injecting**	**ECCS Throttled**	**# Loops of SPC**	**LP ECCS Pumps Injecting**	**ECCS Throttled**	**# Loops of SPC**	**Time COP Required**	**Time of "Flow Collapse"**	**Comment**
#1	DBA-LOCA	LPCI Loop Select Logic	6 (4 LPCI, 2 CS)	No	0	n/a	n/a	n/a	t=420 sec (7 min.)	t=10 min. (judged conservative)	•Throttling LP ECCS prior to t=~10 min. will restore adequate NPSH •Scenario #1 and #3 can be modeled together as need for COP occurs at approximately same time, throttling will preclude need, and whether or not LPCI loop select logic fails does not impact this result
#2	DBA-LOCA	One Division Emergency AC	3 (2 LPCI, 1 CS)	No	0	1 (1 CS)	Yes	1 (1 RHR pump, 1 Hx, 1 RHRSW pump)	t=8160 sec (136 min.)	t=13560 s (226 min.).	•Need for COP occurs in late time frame •LP ECCS already throttled (i.e., throttling LP ECCS does not preclude need for COP)
#3	DBA-LOCA	Containment Isolation	6 (4 LPCI, 2 CS)	No	0	2 (2 CS)	Yes	2 (2 RHR pumps per loop, 1 Hx per loop, 2 RHR Service Water (RHRSW) pumps per Hx)	t=440 sec (7.3 min.)	t=~10 min. (judged conservative)	•Throttling LP ECCS prior to t=10 min. will restore adequate NPSH •Scenario #1 and #3 can be modeled together as need for COP occurs at approximately same time, throttling will preclude need, and whether or not LPCI loop select logic fails does not impact this result

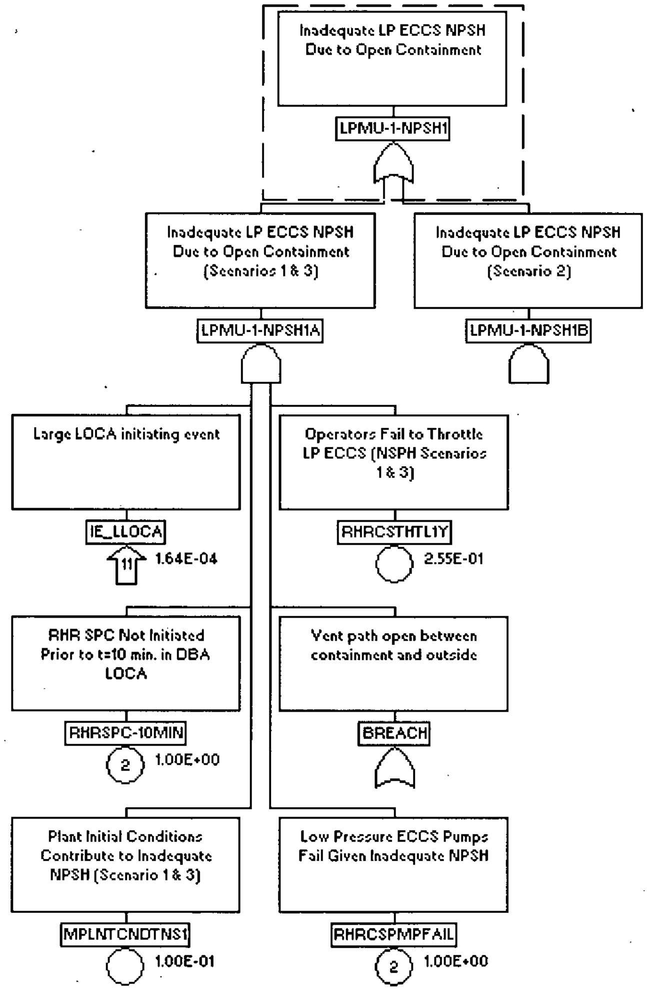

Figure 5-1
DBA-LOCA Scenario Modifications Made to Example Plant PRA

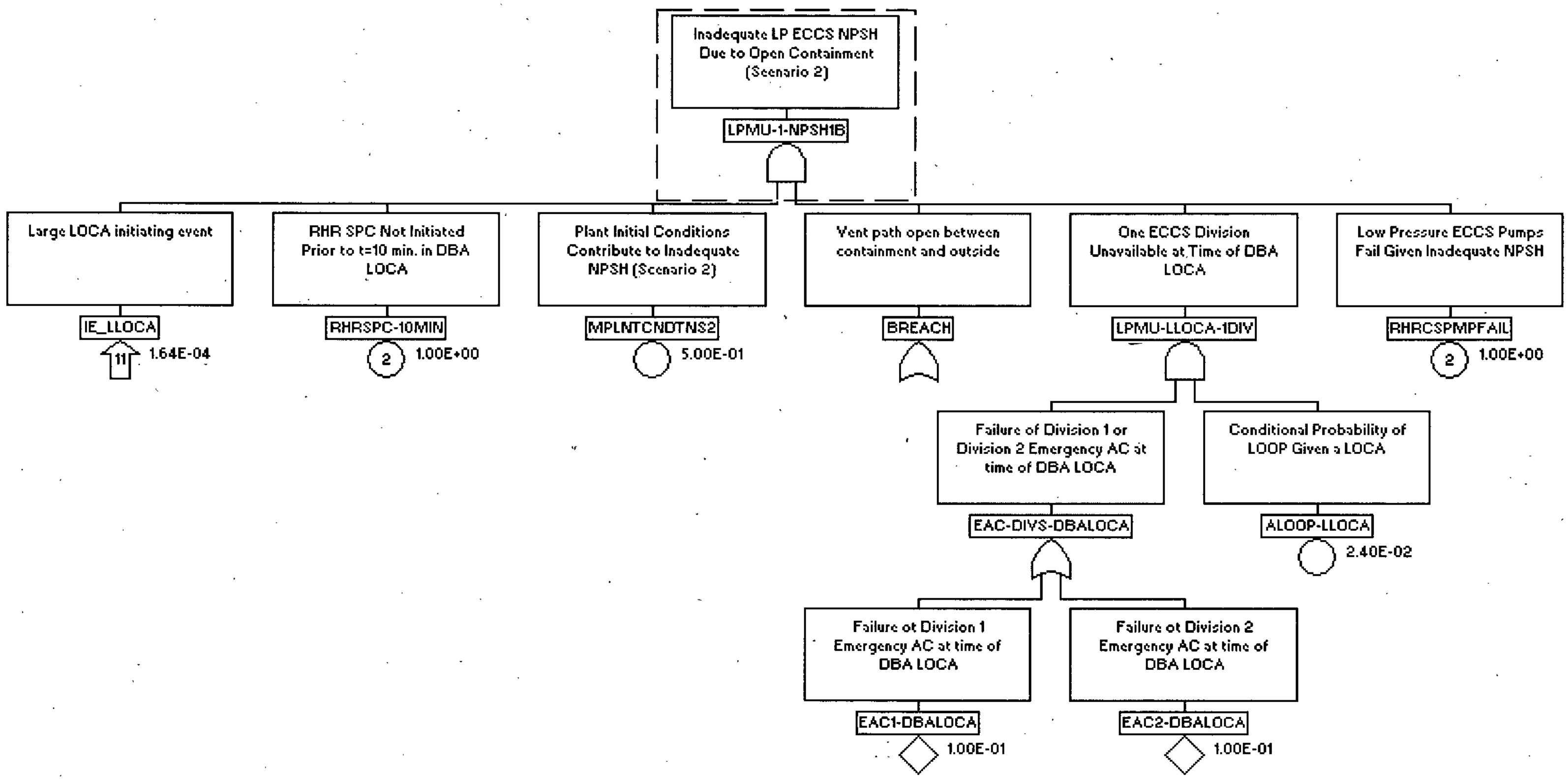

Figure 5-2
DBA-LOCA Scenario Modifications Made to Example Plant PRA

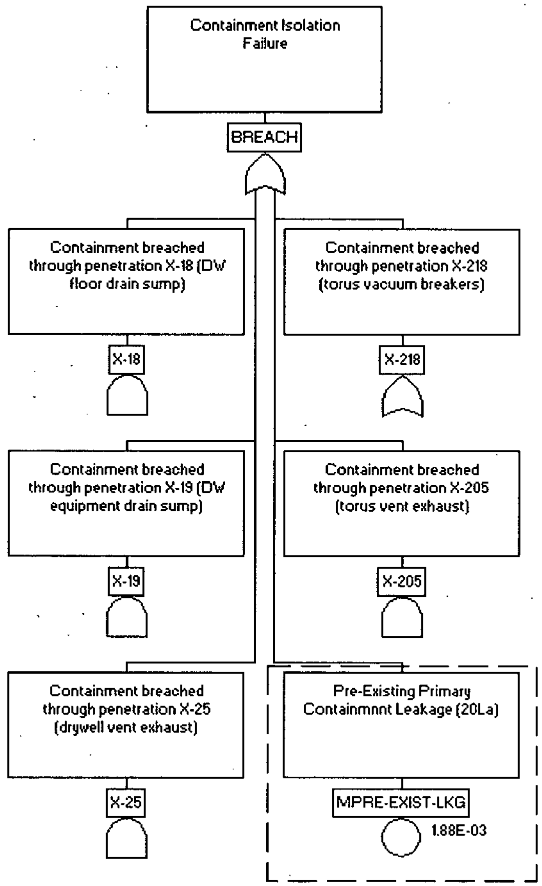

Figure 5-3
DBA-LOCA Scenario Modifications Made to Example Plant PRA

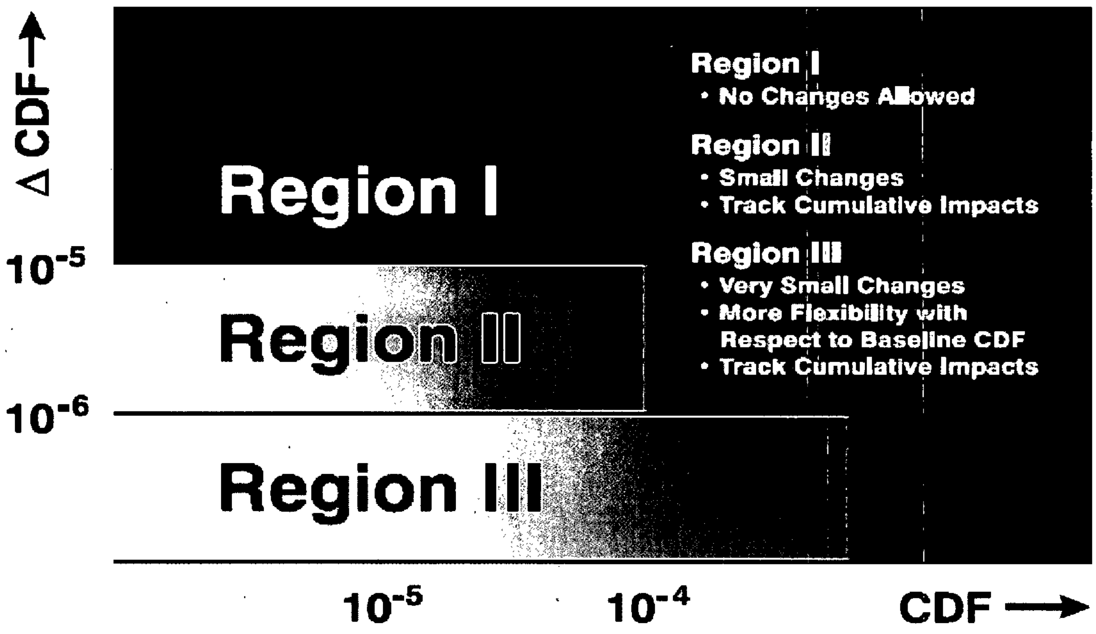

Figure 5-4
RG 1.174 Delta CDF Risk Acceptance Guidelines

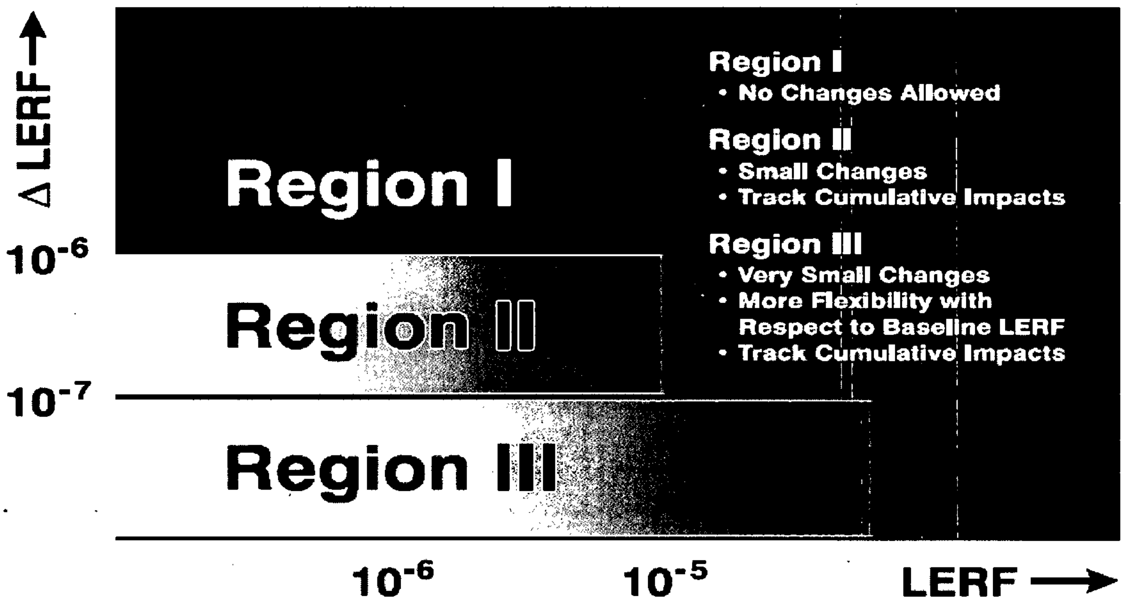

Figure 5-5
RG 1.174 Delta LERF Risk Acceptance Guidelines

6.0 LICENSING BASIS METHODOLOGY & ACCEPTANCE CRITERIA FOR NPSH EVALUATION

6.1 METHODOLOGY FOR CONTAINMENT NPSH ANALYSIS

The following methodology is proposed to address issues associated with the NPSH for pumps that draw from the suppression pool during postulated events.

1. Calculate $NPSH_a$ without COP (deterministic), conservative assumptions, for DBA LOCA and Special Events
2. Action if $NPSH_a < NPSH_r$ (deterministic approach without COP)
3. Action if $NPSH_a > NPSH_r$ (statistical approach without COP)
4. Action if $NPSH_a > NPSH_r$ (statistical approach with COP)
5. Action if $NPSH_a < NPSH_r$ (statistical approach with COP)
6. Validate the generic risk assessment in Section 5.3 or perform plant-specific risk assessment

Details for each of these steps are supplied below.

1. The $NPSH_a$, without COP, is calculated for the DBA-LOCA, using the deterministic approach with conservative assumptions, as described in Section 3.0, such that the resulting suppression pool temperature response is higher while the wetwell pressure response is lower. For special (beyond design basis) events such as Appendix R, SBO and ATWS, the $NPSH_a$ is evaluated in a similar manner, also using the deterministic approach, as described in Section 4.0. Curves are produced showing the amount of COP required and duration compared to COP available to determine licensing basis containment overpressure credit. If $NPSH_a$ with deterministic COP is less than $NPSH_r$, steps 2 or 5 above must be addressed as appropriate.
    a. Acceptance Criterion: The calculated deterministic wetwell pressure (P_{ww}) should be greater than the wetwell pressure required for adequate NPSH such that the wetwell pressure credit granted in the licensing basis minimizes the likelihood of having to seek additional ad hoc regulatory relief.
    b. Acceptance criterion: Wetwell pressure should be shown to be available for the duration required to assure adequate $NPSH_a$, with at least the allowable containment leakage per the Technical Specifications taken into account. (A possible duration may be the maximum coping time after which assured cooling water sources other than the suppression pool will be available for reactor core cooling and containment cooling.)
2. If the $NPSH_a$, without COP, based on the deterministic approach is found to be lower than the $NPSH_r$, then the $NPSH_a$, with COP included, is evaluated using a statistical approach as outlined in Step 3.
3. If $NPSH_a$ is greater than $NPSH_r$ using the statistical approach without COP included, then credit for deterministic COP from Step 1 is justified on the basis that COP is not realistically needed.

4. If $NPSH_a$ is greater than $NPSH_r$ using the statistical approach with COP included, then credit for COP is justified based on assurance of available COP under expected conditions.
    a. Acceptance Criterion, DBA LOCA: The calculated 95/95 wetwell pressure (P_{ww}) should be greater than the wetwell pressure required for adequate NPSH such that the wetwell pressure credit granted in the licensing basis minimizes the likelihood of having to seek additional ad hoc regulatory relief.
    b. Acceptance Criterion, Special Events: The calculated mean wetwell pressure (P_{ww}) should be greater than the wetwell pressure required for adequate NPSH such that the wetwell pressure credit granted in the licensing basis minimizes the likelihood of having to seek additional ad hoc regulatory relief
    c. Acceptance Criterion, Special Events: Wetwell pressure should be shown to be available for the duration required to assure adequate $NPSH_a$, with at least the allowable containment leakage per the Technical Specifications taken into account. (A possible duration may be the maximum coping time after which assured cooling water sources other than the suppression pool will be available for reactor core cooling and containment cooling.)
5. If the $NPSH_a$, with COP included, based on the statistical approach is still found to be lower than the $NPSH_r$, then it will be necessary to demonstrate that operation of the ECCS pumps is acceptable by presenting alternative methods (e.g., plant procedures to monitor pump performance, addressing pump operational capability at degraded conditions as described in Section 5.1 as needed). This is discussed further in Section 5.2 (Alternate Methods to COP) for evaluating the need for COP to satisfy the NPSH concerns of ECCS pumps taking suction from the suppression pool during postulated events.
6. Validate the generic risk assessment in Section 5.3 or perform plant-specific risk assessment

The methodology above is applied with the level of detail as outlined in Appendix A as part of the licensing submittal for COP. Plant-specific $NPSH_a$ calculation results using Appendix A as an example are to be included with the license submittal. The description of the risk assessment and the parameters used should be compared to the applicant's PRA. If the risk assessed is consistent with the applicant's PRA, a justification should be provided. If not, a plant-specific risk assessment should be performed.

In granting approval of COP in the license, NRC should provide a statement consistent with the plant submittal for items 1 through 5 as applicable, for example that $NPSH_a > NPSH_r$ for short term and long term, using the deterministic or statistical method, using this topical report approved methodology, including approval of throttling and cavitation as needed. Specific time-dependent COP will not need to appear in the license since an NRC approved methodology was used. This allows for corrective action program corrections not to require LARs should minor changes to the calculations be needed, which would not change the conclusion or element (1 through 5) of the methods approved.

6.2 METHODOLOGY ELEMENTS DEFINING THE LICENSING BASIS

The methodology described in Section 6.1 proposes to use the deterministic LOCA and special events analysis cases with their associated conservative inputs as the licensing basis for containment overpressure. While the industry trend is toward increasing use of risk information, the use of risk information is less appropriate for supporting the design basis in the containment overpressure analysis licensing area.

The primary reason for this is the high workload required for the configuration management of the statistical approach. 10CFR50 Appendix B requirements that apply to design basis calculations require configuration control. The requirement is to maintain current inputs and associated results for ongoing configuration management. The Monte Carlo (statistical) approach utilizes several years of plant data to develop exceedance probabilities and 59 separate calculations are required to achieve a statistically significant result. The continuously varying plant conditions that provide input to these calculations would therefore require frequent repetition of the calculations to assure that design bases were not exceeded. Therefore, it is not practical to use the Monte Carlo (statistical) approach, with exceedance probabilities for each of the variable inputs, to continuously re-calculate the statistical exceedance probabilities and the resulting impact. The deterministic method calculations, though more conservative, would not require this type of re-evaluation when plant parameters change.

By utilizing the Monte Carlo calculations only as another level of evaluations to demonstrate the margin inherent in the deterministic method, the licensing basis is maintained without excessive configuration management requirements.

7.0 ELEMENTS OF LICENSE AMENDMENT REQUEST FOR CONTAINMENT OVERPRESSURE CREDIT

Licensees will develop a Licensing Amendment Request (LAR) using their standard LAR format. The LAR will contain the following information in the Technical Evaluation:

TECHNICAL EVALUATION

- NPSH Evaluation of Special Events (e.g., Appendix R, SBO, ATWS)
- Risk Assessment
- Available NPSH Evaluation for DBA-LOCA
- Generic issues addressed and approved by topical report
- Plant-specific topics to be addressed
- Explain why cannot be addressed generically.
- Exceptions

8.0 REFERENCES

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Appendix A
DBA-LOCA Containment Response Evaluation for Use in NPSH Evaluation for Monticello Nuclear Generating Plant

A DBA-LOCA containment analysis is performed for the MNGP, as an example of its application, using the deterministic and statistical approach described in Section 3.1. The $NPSH_a$ is calculated, using the results of this containment analysis, and the results are compared with the $NPSH_r$. For this analysis, the plant is assumed to be at EPU conditions (i.e. 120% of Original Licensed Thermal Power). Table A-1 shows key plant conditions assumed for the deterministic approach. The short-term and long-term DBA-LOCA scenarios analyzed are described below.

Short-term DBA-LOCA – LPCI loop selection logic failure as single failure

- A double-ended guillotine break of a recirculation line (DBA-LOCA) occurs at time zero.
- The analysis time of interest is up to 10 minutes into the event before the RHR containment cooling is initiated.
- All four LPCI pumps and two low-pressure core spray (LPCS) pumps are operating according to automatic initiation signals.
- With the assumed failure of loop selection logic, all of the LPCI flow goes into a broken recirculation line.
- The LPCI flow injecting into the broken line is discharged directly into the drywell.
- Runout flow occurs for ECCS pumps as the vessel is depressurized.

Long-term DBA-LOCA – diesel generator failure as single failure

- A double-ended guillotine break of a recirculation line (DBA-LOCA) occurs at time zero.
- The analysis time of interest is after 10 minutes into the event after the RHR pool cooling is initiated.
- With the assumed failure of one diesel generator, one LPCS pump and two LPCI pumps are operating during the first 10 minutes into the event. At 10 minutes, one LPCI pump flow is realigned in containment spray cooling mode, and the other LPCI pump is turned off to activate one RHR service water pump, and the operator throttles the flow from ECCS pumps.

The above two scenarios are analyzed using the deterministic approach with conservative assumptions, and also using the statistical approach.

A.1 Deterministic Evaluation of Containment Response with Conservative Assumptions

The deterministic approach uses the following key conservative input assumptions that result in a higher pool temperature response:

- Initial reactor power level at 102% rated.

- Nominal decay heat plus 2-sigma uncertainty after reactor SCRAM
- Maximum initial suppression pool temperature
- Maximum service water (ultimate heat sink) temperature
- Minimum RHR heat exchanger heat removal capability
- Minimum initial suppression pool volume
- Maximum initial drywell temperature
- Minimum initial drywell pressure
- Minimum initial wetwell pressure
- 100% relative humidity for both drywell and wetwell airspace
- Minimum ambient pressure
- Technical Specification containment allowed leakage rate
- The portion of feedwater inventory initially above 212°F flows into the reactor vessel after absorbing heat from the feedwater pipe metal as it flows toward the vessel.
- [[

  ]]
- The operator initiates suppression pool cooling in containment spray mode (rather than direct pool cooling) at 10 minutes into the event, and the spray droplets are in thermal equilibrium with the containment airspace before falling on the bottom of drywell or the suppression pool.

Table A-1 provides the values of key input parameters used in the deterministic approach. Using these input values, the GEH containment analysis code SHEX is run for the two scenarios of the DBA-LOCA identified above. For each scenario, the values of H_{ww} (as defined in Section 2.0) are obtained as a function of time. These values are plotted in Figures A-1 (Short-term), and A-2 (Long-term), along with the results from the statistical approach explained below.

A.2 Statistical Evaluation of Containment Response with Realistic Input Assumptions

In the statistical approach, the following input parameters are statistically defined:

- Initial reactor power level.
- Decay heat value after reactor SCRAM
- Initial suppression pool temperature
- Service water (ultimate heat sink) temperature
- RHR heat exchanger heat removal capability
- Initial suppression pool volume
- Initial drywell temperature
- Initial drywell pressure
- Minimum initial wetwell pressure

- Technical Specification containment allowed leakage rate

The input parameters identified above are varied randomly based on their probability distribution. For other input parameters (not statistically defined), the input assumptions are the same as those for the deterministic approach. For instance, the statistical approach also conservatively assumes 100% relative humidity for both the drywell and wetwell airspace, and 14.26 psia for the pressure in the reactor building. A total of 59 sets of input values are generated randomly (i.e., from the corresponding probability distributions) for each input parameter according to the Monte Carlo process to obtain a statistically meaningful number of trials.

Probability Distribution of Statistically Defined Input Parameters

The plant operating parameters, such as power level, suppression pool temperature, service water temperature, etc. are measured periodically at the MNGP. The plant data processed to derive a probability distribution for these parameters. The probability distributions based on measurement data are then used to determine the 59 sets of input values, as explained below.

As an example, the suppression pool temperature measurement data can be sorted into 5-degree temperature bins, showing the number of days (frequency) for each temperature bin. The results of sorting are given in Table A-2. Once the measurement data are sorted, the probability of exceedance for a given value is determined. The exceedance probability represents the relative frequency when the operating parameter is above a certain value, and its value is calculated by dividing the number of days when the operating parameter is above a given value by the total number of days. For example, the exceedance probability for the minimum value is 1.0, whereas the exceedance probability for the maximum value is zero. Table A-3 shows the exceedance probability for the suppression pool temperature, which is generated from Table A-2. Likewise, the measurement data of other parameters are sorted from Tables A-4, A-6, A-8, A-10, A-12, A-14 and A-16 and the exceedance probability for these parameters is generated and shown in Tables A-5, A-7, A-9, A-11, A-13, A-15 and A-17, respectively.

Once the exceedance probability table is generated, 59 random values are generated from a uniform distribution between 0 and 1.0. Then, 59 random values of the parameter are determined by matching a random value from the uniform distribution with an exceedance probability value. In this way, about 2% of the suppression pool temperature random values will fall between 85°F and 90°F in the above example (see Table A-2), thus representing the profile of the suppression pool temperature. If the measurement data for a given parameter are skewed, more random values will be selected from that side, representing the profile of that parameter.

The plant operates at reactor thermal power level not exceeding its licensed power level. The reactor thermal power is a derived value, calculated using measured values of reactor operating parameters, such as feedwater flow rate and enthalpy, etc. The accuracy of these measurements determines the uncertainty in the power level calculation. Because of this uncertainty, 2% is added to the rated power in safety-related analyses.

Containment leakage tests are performed at power plants (per 10CFR Appendix J to Part 50 requirements) to ensure that the containment leakage rate is less than the value specified in the Technical Specification. Individual power plants provide a very limited data set beyond the Technical Specification allowed leakage rate. Because of a lack of plant specific data, a compilation of industry-wide containment leakage data as provided in an EPRI report (Reference

A.5-1) is used. From the compilation of the leakage events, the report assessed the likelihood of pre-existing containment leakages, and summarized the resulting probabilities as a function of pre-existing leakage sizes. For this statistical analysis for the MNGP, the EPRI results are used after augmenting the probability by a factor of 2. Tables A-4 and A-5 show the EPRI results and the probability values used for this analysis, respectively. For statistical analysis, the lowest containment leakage rate probabilities to be used are the EPRI probability values from Table A-4. The random values for this parameter are determined in a manner similar to that for the measurement data explained above.

A decay heat table is generated for the MNGP, based upon the American National Standards Institute (ANSI)/American Nuclear Society (ANS) 5.1-1979 decay heat model. The decay heat calculation provides nominal values and one-sigma uncertainty values (as percentage of nominal value) as a function of time after reactor SCRAM. A normal distribution is assumed for the reactor thermal power and for the decay heat. The standard deviation values for these normal distributions are assumed to be 1% for the reactor power and one-sigma value for the decay heat. Accordingly, random values are generated based on a normal distribution. Since the decay heat uncertainty is calculated as a function of time, random values for the decay heat are determined as a function of time.

Thus, 59 sets of random values are generated for each of the input parameters identified above, as input to the GEH containment analysis code SHEX to obtain 59 SHEX cases for the DBA-LOCA. Using these input values, SHEX is run for the short-term and long-term DBA-LOCA scenarios. This produces the SHEX results for the 59 cases for each of the two time domains. For each scenario, the minimum value of H_{ww}, as its 95/95 value, is determined as a function of time from the 59 SHEX runs. Here, the reference water level used to calculate the value of H_{ww} (see Equation (2)) is the elevation of the suppression pool surface at the minimum pool volume (68,000 ft^3).

The 95/95 value of H_{ww} obtained from the statistical approach is compared with the value obtained from the deterministic approach in Figures A-1 and A-2 for the short-term and long-term DBA-LOCA, respectively. The H_{ww} values obtained with the deterministic approach bound (i.e., are lower than) the 95/95 values (minimum values) calculated by the statistical approach. This indicates that the deterministic approach with conservative assumptions will result in a conservative estimate of the $NPSH_a$, compared with the 95/95 value based on the statistical approach. Later in this appendix, the $NPSH_a$ is calculated by adding H_{pl} to H_{ww} (see Equation 2 in Section 2.2), and the available NPSH margin ($NPSH_a - NPSH_r$) is calculated.

The suppression pool temperature response is also compared in Figures A-3 and A-4 between the deterministic approach and the statistical approach. As expected, the results show that the deterministic approach results in higher suppression pool temperatures, compared with the 95/95 values calculated by the statistical approach.

A.3 Evaluation of $NPSH_a$

Sections A.1 and A.2 provide the value of H_{ww} (see Section 2.2 for a definition of this parameter) from the containment response evaluations based on the deterministic and statistical method, respectively. The $NPSH_a$ is calculated as a function of time by adding H_{pl} (see Equation 2 in Section 2.2) to H_{ww} and comparing it with the $NPSH_r$. The value of H_{pl} is calculated as:

$H_{pl} = H_{pool} - H_{pump} - H_{loss}$

In the above equation, the value of (H_{pool} - H_{pump}) is an elevation difference (ft) between the pool elevation (the pool surface at the minimum initial pool volume of 68,000 ft^3) and the pump inlet. The value of H_{loss} (suction strainer and suction line head loss) is calculated according to the method described in Section 3.2. Specifically, the following steps were taken:

1. Flow rates and flow velocities through the four suction strainer lines were determined for the event initial conditions with a fixed K value for clean suction strainers in both the short term and long term scenarios.
2. Using the flow rates and flow velocities determined, the amount of debris that collects on each suction strainer and the new fixed K value for each strainer were calculated. The specific time interval used and suppression pool temperature at the end of that interval were provided by the containment accident analyses (Sections A.1 and A.2) for the particular accident scenario being evaluated.
3. New flow rates and flow velocities through the four suction strainers were established for the new debris loading conditions.
4. Steps 2 and 3 were repeated as required until the debris loading is reasonably consistent with the flow rates and velocities.
5. Based on the debris loading and flow rates and velocities determined in Step 4, the value of H_{pl} is calculated.

There are six ECCS pumps operating for the short-term DBA-LOCA scenario: 2 CS pumps and 4 LPCI/RHR pumps. The value of H_{pl} is calculated for each of the six pumps. Note that there are differences in H_{pl} between CS pumps and also between LPCI/RHR pumps, since there are some differences in the piping configuration connected to the pumps. For the long-term DBA-LOCA, two ECCS pumps (one CS pump and one LPCI/RHR pump) are operating after 600 seconds due to failure of one of the two diesel generators (A and B), and H_{pl} is calculated for each of the two failure scenarios: failure of diesel generator A and failure of diesel generator B.

After calculating H_{pl}, the value of $NPSH_a$ is calculated as follows:

1. For the deterministic approach, $NPSH_a$ is calculated as a function of time by adding H_{pl} to H_{ww} calculated in Section A.1
2. For the statistical approach, the 95/95 value of $NPSH_a$ is calculated as a function of time by adding H_{pl} to the minimum value of H_{ww} that is calculated as a function of time in Section A.2.

Thus, $NPSH_a$ is calculated as a function of time for pumps assumed to be operating during the short-term and long-term DBA-LOCA scenarios, and this value is compared with $NPSH_r$. Table A-18 shows the CS pump results for the short-term DBA-LOCA. Here, the worst-case results between the two CS pumps are tabulated. Table A-19 shows the worst-case LPCI pumps results. Tables A-18 and A-19 also show the available wetwell pressure corresponding to $NPSH_a$ in comparison with the required wetwell pressure corresponding to $NPSH_r$. The worst-case results of $NPSH_a$ are also compared with $NPSH_r$ in Figures A-5 and A-6 for the CS and LPCI pumps, respectively. Figures A-9 and A-10 show the pressure results for the short-term DBA-LOCA.

For the long-term DBA-LOCA, the worst-case results for the CS and LPCI/RHR pumps are given in Tables A-20 and A-21, respectively. Figures A-7 and A-8 show the results of $NPSH_a$ for the CS and LPCI/RHR pump, while the wetwell pressure results for the long-term DBA-LOCA are plotted in Figures A-11 and A-12.

Examination of the NPSH evaluation results for the DBA-LOCA presented in this appendix reveals that:

1. Based on the deterministic approach, the $NPSH_a$ is larger than the $NPSH_r$ throughout the event for both the short-term and long-term DBA-LOCA.

2. The calculated wetwell pressure based on the deterministic approach becomes less than the ambient pressure into the event due to the leakage assumption. However, this value is larger than the required wetwell pressure because the suppression pool temperature has already decreased below a value corresponding to the required NPSH after that time.

3. The NPSH margin to the $NPSH_r$, based on the statistical approach, is significantly larger than the NPSH margin obtained with the deterministic approach.

A.4 Conclusion

Based on the proposed methodology and acceptance criteria for the NPSH evaluation, it is concluded that for the MNGP at EPU conditions adequate NPSH exists for ECCS/RHR pumps during the DBA-LOCA.

A.5 References for Appendix A

[1] "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," EPRI Report No. 1009325, Revision 1, October 2005.

Table A-1
Deterministic Approach Input Values for MNGP DBA-LOCA Containment Analysis

Input Parameter	Unit	Value Used in Analysis	Remarks
Initial Reactor Thermal Power	MWt	2004	Rated + 2% (EPU Conditions)
Decay Heat		Nominal + 2 sigma uncertainty	Based on ANSI/ANS 5.1-1979 decay heat model
Initial Suppression Pool Temperature	°F	90	Technical Specification maximum normal operating temperature.
Service Water Temperature	°F	90	Maximum license bases UHS temperature
RHR Heat Exchanger K-value[1)]	Btu/sec- °F	147	At 85°F service water temperature and 125°F pool temperature
Initial Suppression Pool Volume	ft^3	68,000	Technical Specification minimum value
Initial Drywell Temperature	°F	135	Technical Specification maximum Drywell average air temperature
Initial Drywell Pressure	psia	14.26	
Initial Wetwell Pressure	psia	14.26	
Relative Humidity for Drywell and Wetwell Airspace	%	100	Maximum value
Ambient Pressure	psia	14.26	Minimum Ambient pressure
Containment Leakage Rate	%/day	1.2	At 10CFR Appendix J to Part 50 test conditions
Time of Containment Spray Initiation	sec	600	

[1] K-value is defined to be the heat removal rate divided by the difference between two inlet temperatures. The K-value increases as the inlet temperatures increase.

Table A-2
MNGP Suppression Pool Temperature Measurement Data

Low (°F)	High (°F)	Number of Days
65	70	6
70	75	168
75	80	801
80	85	790
85	90	33
90	95	1
	Total	1799

Table A-3
MNGP Suppression Pool Temperature Exceedance Probability Based on Measurement Data

Temperature (°F)	Exceedance Probability
65	1
70	0.997
75	0.903
80	0.458
85	0.0189
90	0.000556
95	0

Table A-4
Probability of Pre-existing Containment Leak from EPRI Study

EPRI Value[2]	
Leak Size (La[3])	**Probability of Occurrence**
1	2.65E-02
2	1.59E-02
5	7.42E-03
10	3.88E-03
20	1.88E-03
35	9.86E-04
50	6.33E-04
100	2.47E-04
200	8.57E-05
500	1.75E-05
600	1.24E-05

Table A-5
Pre-existing Containment Leak Exceedance Probability Assumed for MNGP Containment Monte Carlo Evaluation

Analysis Value (2 times EPRI Probability Value)	
Leak Size (La)	**Exceedance Probability**
1	1 to 5.30E-02
2	3.18E-02
5	1.48E-02
10	7.76E-03
20	3.76E-03
35	1.97E-03
50	1.27E-03
100	4.94E-04
200	1.71E-04
500	3.50E-05
600	2.48E-05

[2] From Reference A.5-1
[3] 1 La is the Technical Specification containment allowed leakage rate

Table A-6
MNGP Service Water Measurement Data

Low (°F)	High (°F)	Number of Days
32	35	603
35	40	133
40	45	74
45	50	96
50	55	131
55	60	136
60	65	78
65	70	155
70	75	218
75	80	146
80	85	27
85	90	1
	Total	1798

Table A-7
MNGP Service Water Exceedance Probability Based on Measurement Data

Temperature (°F)	Exceedance Probability
32	1
35	0.664627
40	0.590656
45	0.549499
50	0.496107
55	0.423248
60	0.347608
65	0.304227
70	0.21802
75	0.096774
80	0.015573
85	0.000556
90	0

Table A-8
MNGP Suppression Pool Volume Measurement Data

SP Volume, Low (ft^3)	SP Volume, High (ft^3)	Number of Days
68000	69383.73	1
69383.73	69830.09	181
69830.09	70276.45	348
70276.45	70722.82	365
70722.82	71169.18	376
71169.18	71615.55	290
71615.55	72061.91	202
72061.91	72508.27	42
72508.27	72954.64	3
	Total	1808

Table A-9
MNGP Suppression Pool Volume Exceedance Probability Based on Measurement Data

SP volume (ft^3)	Exceedance Probability
68000	1
69383.73	0.999447
69830.09	0.899336
70276.45	0.706858
70722.82	0.504978
71169.18	0.297013
71615.55	0.136615
72061.91	0.024889
72508.27	0.001659
72954.64	0

Table A-10
MNGP Drywell Temperature Measurement Data

Low (oF)	High (oF)	Number of Days
72.5	77.5	2
77.5	82.5	19
82.5	87.5	36
87.5	92.5	17
92.5	97.5	8
97.5	102.5	10
102.5	107.5	61
107.5	112.5	1159
112.5	117.5	468
117.5	122.5	37
122.5	127.5	0
	Total	1817

Table A-11
MNGP Drywell Temperature Exceedance Probability Based on Measurement Data

Temperature (°F)	Exceedance Probability
72.5	1
77.5	0.998899
82.5	0.988442
87.5	0.96863
92.5	0.959274
97.5	0.954871
102.5	0.949367
107.5	0.915795
112.5	0.277931
117.5	0.020363
122.5	0

Table A-12
MNGP Drywell Pressure Measurement Data

Low (psig)	High (psig)	Number of Days
-0.25	0.00	0
0.00	0.25	112
0.25	0.50	975
0.50	0.75	648
0.75	1.00	42
1.00	1.25	2
1.25	1.50	0
	Total	1779

Table A-13
MNGP Drywell Pressure Exceedance Probability Based on Measurement Data

Pressure (psig)	Exceedance Probability
0.00	1
0.25	0.937043
0.50	0.388983
0.75	0.024733
1.00	0.001124
1.25	0
1.50	0

Table A-14
MNGP Wetwell Pressure Measurement Data

Low (psig)	High (psig)	Number of Days
-0.25	0.00	0
0.00	0.25	95
0.25	0.50	913
0.50	0.75	734
0.75	1.00	44
1.00	1.25	2
1.25	1.50	0
	Total	1788

Table A-15
MNGP Wetwell Pressure Exceedance Probability Based on Measurement Data

Pressure (psig)	Exceedance Probability
0.00	1
0.25	0.946868
0.50	0.436242
0.75	0.025727
1.00	0.001119
1.25	0
1.50	0

Table A-16
MNGP RHR K Measurement Data

Low (BTU/sec-°F)	High (BTU/sec-°F)	Number of Days
132.5	137.5	0
137.5	142.5	0
142.5	147.5	1
147.5	152.5	4
152.5	157.5	6
157.5	162.5	4
162.5	167.5	2
167.5	172.5	1
172.5	177.5	2
177.5	182.5	0
	Total	20

Note: With RHR temperature < 80 °F and RHR service water < 45 °F

Table A-17
MNGP RHR K Exceedance Probability
Based on Measurement Data

RHR K (BTU/sec-°F)	Exceedance Probability
142.5	1
147.5	0.95
152.5	0.75
157.5	0.45
162.5	0.25
167.5	0.15
172.5	0.1
177.5	0

Table A-18
Results of CS Pump NPSH Evaluation for the Short-Term DBA-LOCA

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required Wetwell (WW) Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
96	67.14		26.6	40.54		-1.16		16.15	
120		70.68	26.6		44.08		-1.41		17.09
185	52.49		26.6	25.89		-0.73		10.31	
282		36.14	26.6		9.54		-0.62		2.61
358	31.14		26.6	4.54		0.43		2.36	
420		32.34	26.6		5.74		0.21		2.11
476	29.43		26.6	2.83		1.11		2.31	
588		31.73	26.6		5.13		1.00		2.55
590	29.28		26.6	2.68		1.56		2.70	

Table A-19
Results of LPCI Pump NPSH Evaluation for the Short-Term DBA-LOCA

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
96	67.93		25.5	42.43		-1.97		16.15	
120		71.47	25.5		45.97		-2.23		17.09
185	53.28		25.5	27.78		-1.53		10.31	
282		36.92	25.5		11.42		-1.42		2.61
358	31.88		25.5	6.38		-0.35		2.36	
420		33.07	25.5		7.57		-0.57		2.11
476	30.16		25.5	4.66		0.33		2.31	
588		32.45	25.5		6.95		0.22		2.55
590	30.00		25.5	4.50		0.79		2.70	

Table A-20
Results of CS Pump NPSH Evaluation for the Long-Term DBA-LOCA

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
600		56.19	23.0		33.19		-4.10		9.54
978	36.27		23.0	13.27		-2.91		2.72	
5558	35.13		23.0	12.13		0.89		5.98	
8160		36.07	23.0		13.07		0.18		3.34
13560		35.86	23.0		12.86		1.24		3.32
21517	33.00		23.0	10.00		5.45		9.62	
25800		36.12	23.0		13.12		1.84		2.79
34748	32.59		23.0	9.59		6.01		10.00	
40440		36.39	23.0		13.39		1.45		1.89
46321	32.80		23.0	9.80		5.72		9.80	
80325	33.27		23.0	10.27		3.56		7.85	
85440		35.84	23.0		12.84		-0.79		0.31
119240		35.36	23.0		12.36		-1.80		-0.06
168240		35.65	23.0		12.65		-2.67		-0.06
233659	33.08		23.0	10.08		-1.53		2.73	
472096	32.56		23.0	9.56		-3.27		0.79	

Table A-21
Results of LPCI/RHR Pump NPSH Evaluation for the Long-Term DBA-LOCA

Time (sec)	NPSH$_a$ (ft)		NPSH$_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
600		57.18	22.00		35.18		-4.94		9.54
978	37.26		22.00	15.26		-3.75		2.72	
8160		37.05	22.00		15.05		-0.66		3.34
5558	36.10		22.00	14.10		0.06		5.98	
13560		36.84	22.00		14.84		0.41		3.32
21517	33.98		22.00	11.98		4.63		9.62	
25800		37.09	22.00		15.09		1.01		2.79
34748	33.57		22.00	11.57		5.18		10.00	
40440		37.37	22.00		15.37		0.62		1.89
46321	33.78		22.00	11.78		4.90		9.80	
80325	34.24		22.00	12.24		2.74		7.85	
85440		36.82	22.00		14.82		-1.63		0.31
119240		36.34	22.00		14.34		-2.64		-0.06
168240		36.63	22.00		14.63		-3.51		-0.06
233659	34.06		22.00	12.06		-2.37		2.73	
472096	33.54		22.00	11.54		-4.11		0.79	

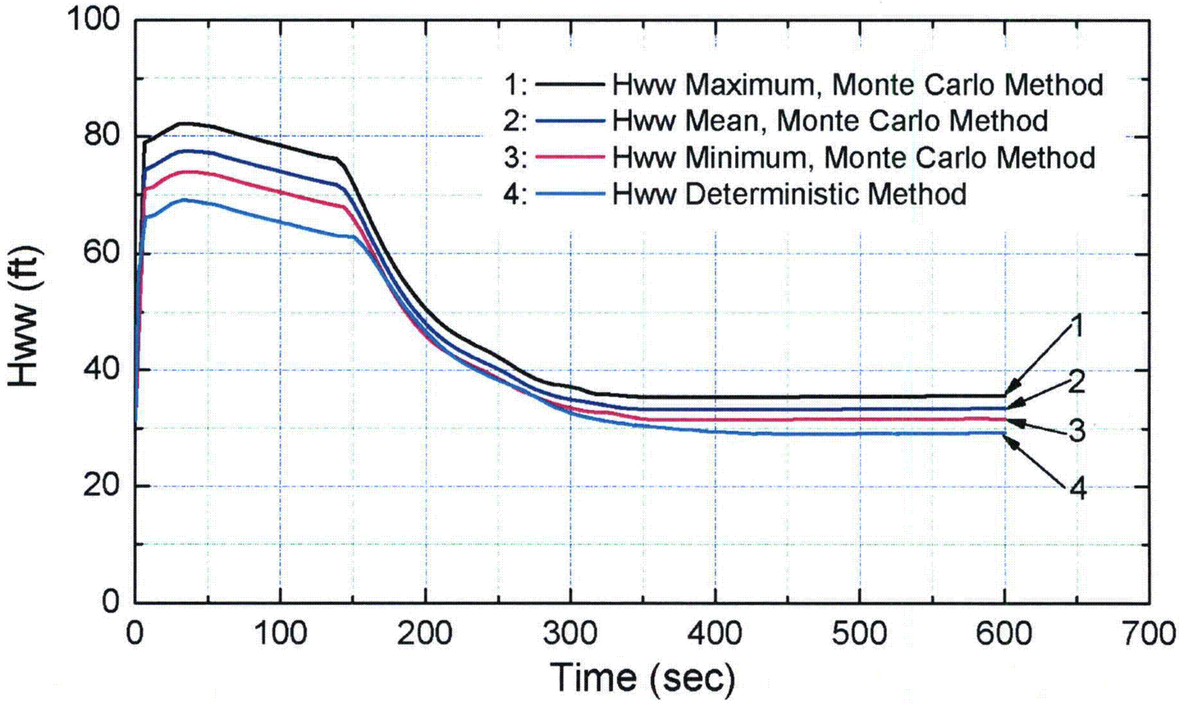

Figure A-1 Comparison of NPSH H_{ww} Values for Short-Term DBA-LOCA (with Loop Selection Logic Failure) between Deterministic Analysis and Statistical Analysis

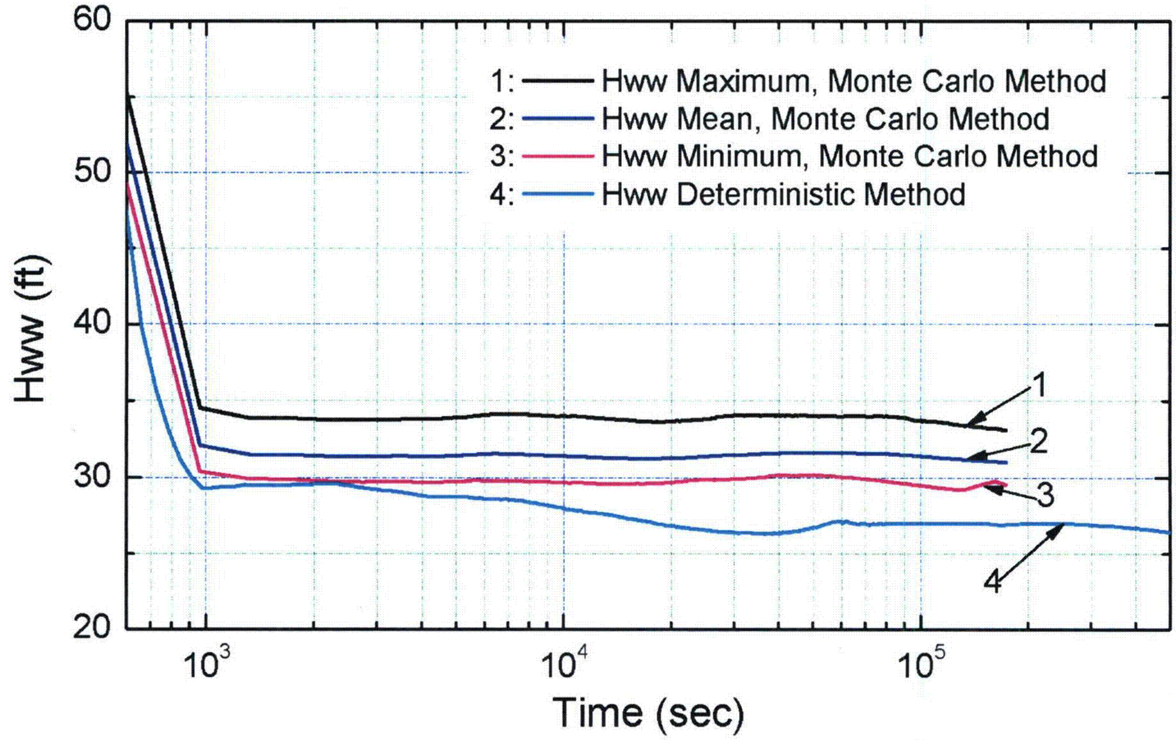

Figure A-2 Comparison of NPSH H_{ww} Values for Long-term DBA-LOCA (with Diesel Generator Failure) between Deterministic Analysis and Statistical Analysis

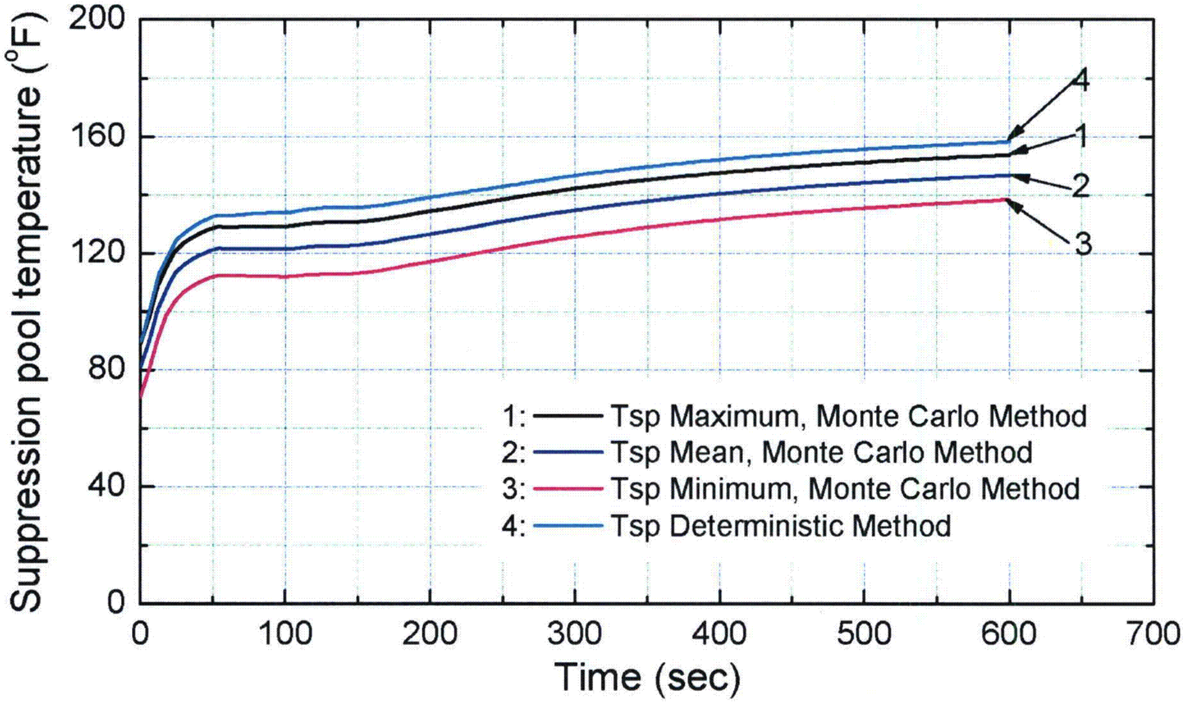

Figure A-3 Comparison of Suppression Pool Temperature for Short-term DBA-LOCA (with Loop Selection Logic Failure) between Deterministic Analysis and Statistical Analysis

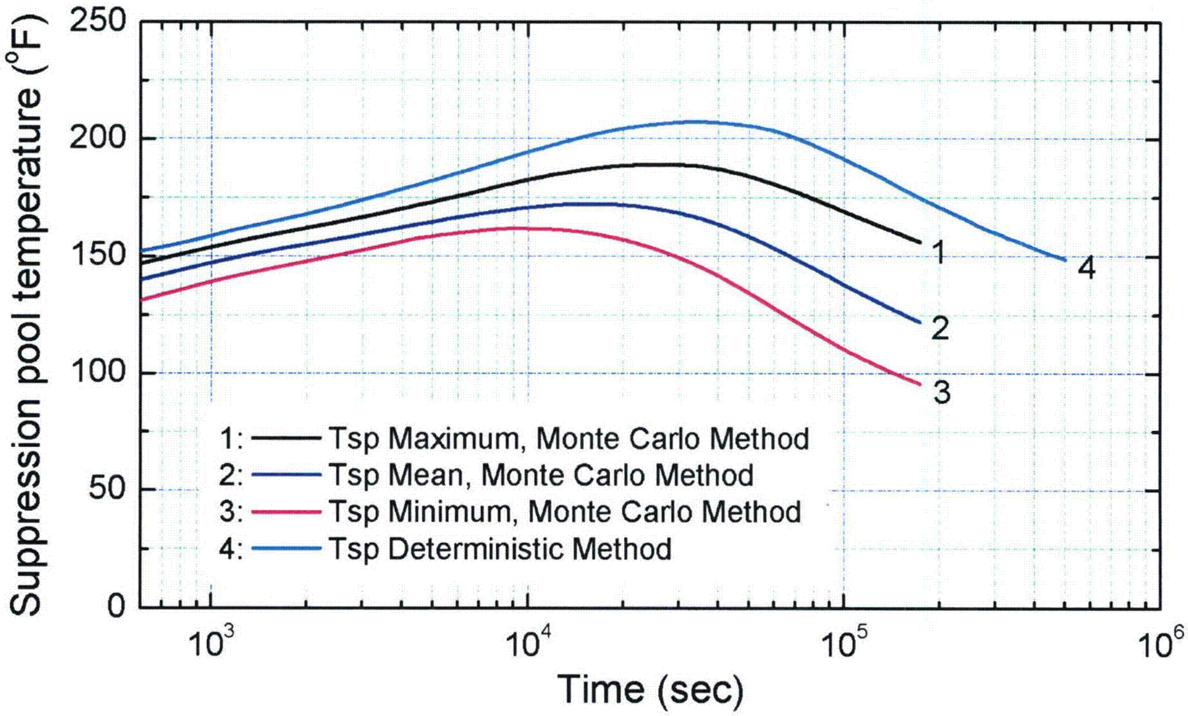

Figure A-4 Comparison of Suppression Pool Temperature for Long-term DBA-LOCA (with Diesel Generator Failure) between Deterministic Analysis and Statistical Analysis

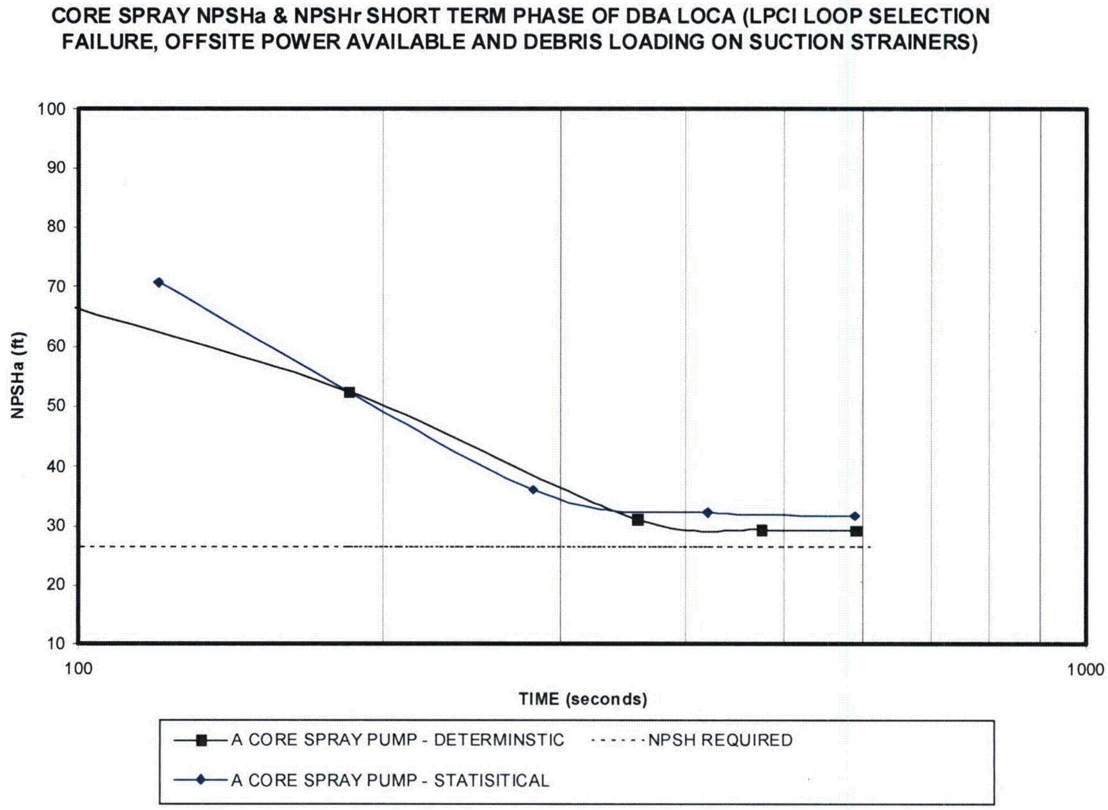

Figure A-5 CS Pump $NPSH_a$ vs. $NPSH_r$ for the Short-term DBA-LOCA

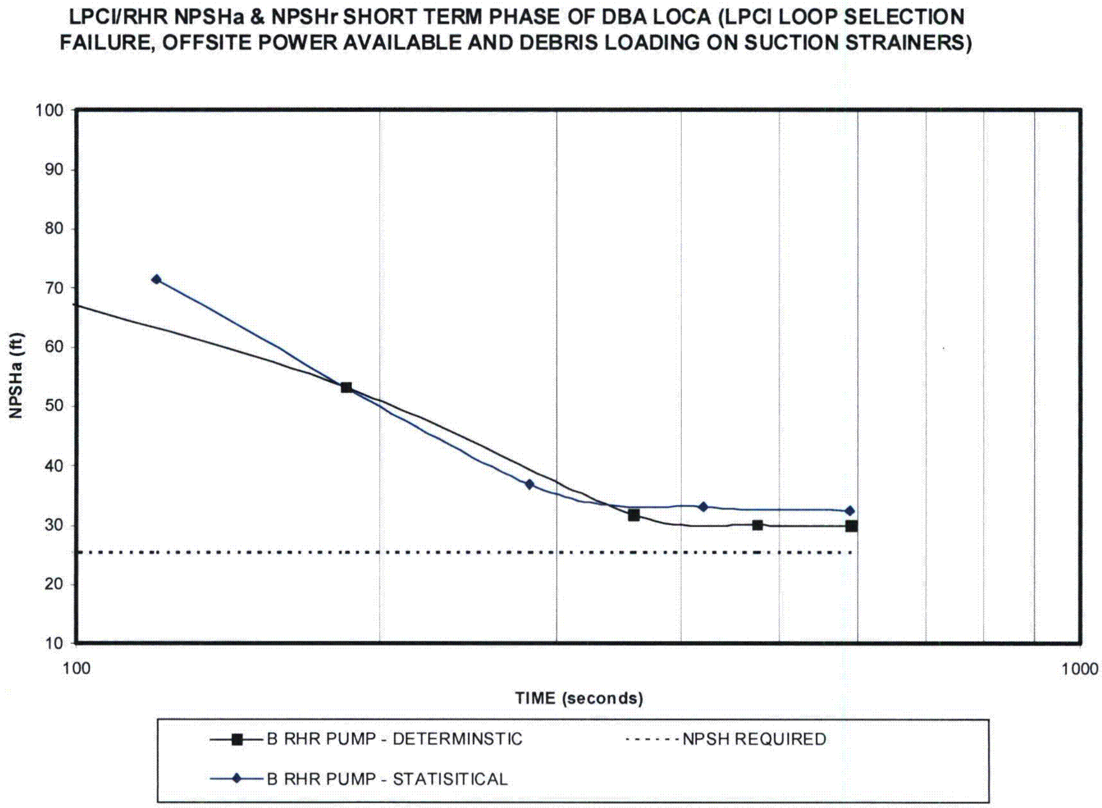

Figure A-6 LPCI Pump $NPSH_a$ vs. $NPSH_r$ for the Short-term DBA-LOCA

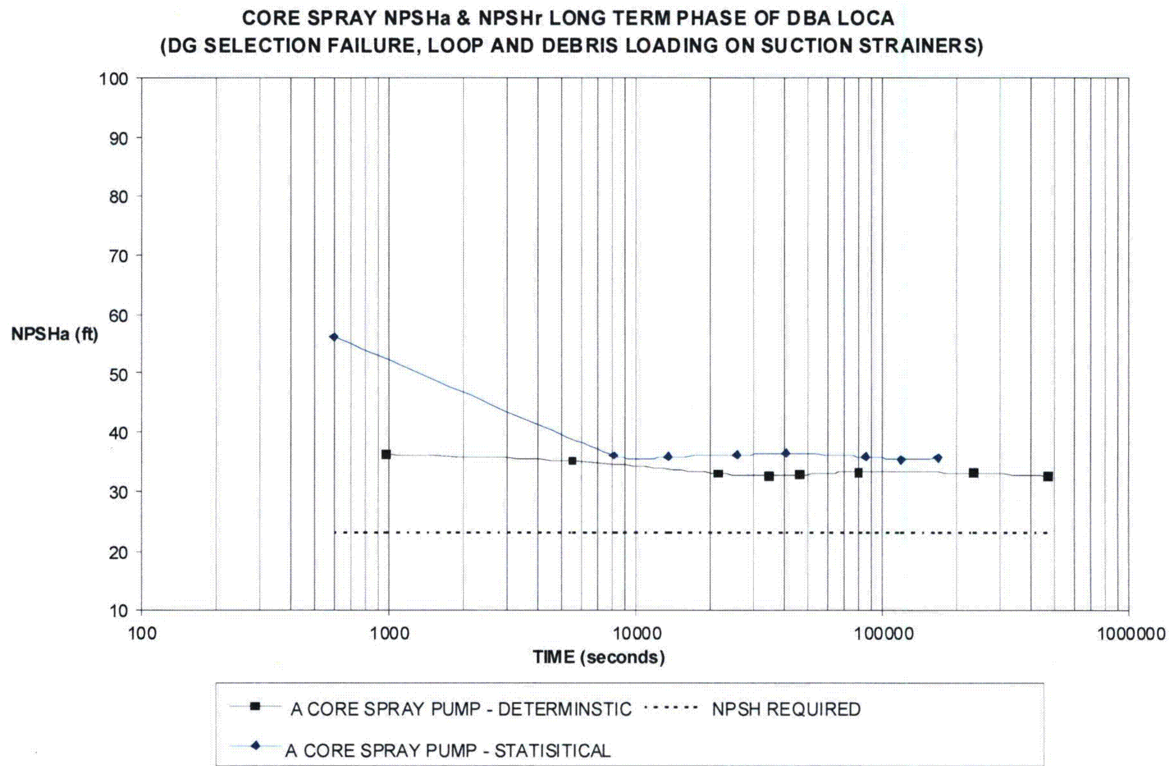

Figure A-7 CS Pump $NPSH_a$ vs. $NPSH_r$ for the Long-term DBA-LOCA

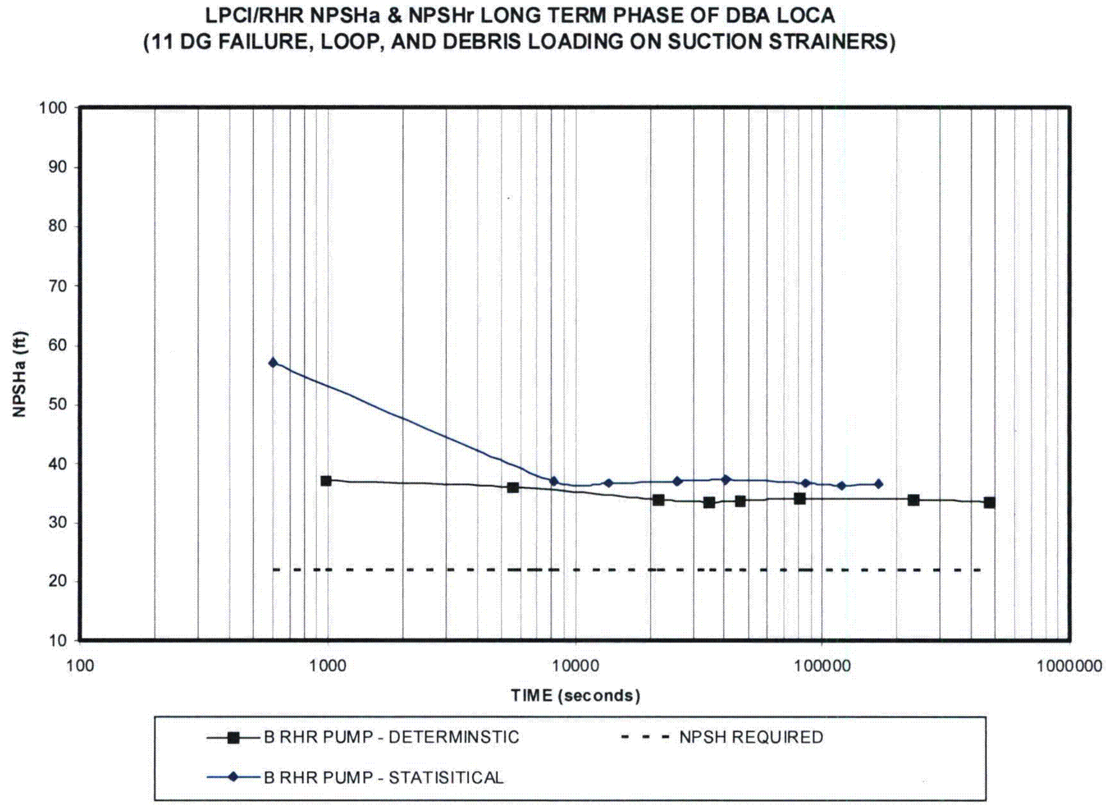

Figure A-8 LPCI Pump $NPSH_a$ vs. $NPSH_r$ for the Long-term DBA-LOCA

CORE SPRAY CONTAINMENT PRESSURE REQUIRED FOR ADEQUATE NPSH DURING THE SHORT TERM PHASE OF DBA LOCA (LPCI LOOP SELECTION FAILURE, OFFSITE POWER AVAILABLE AND DEBRIS LOADING ON SUCTION STRAINERS)

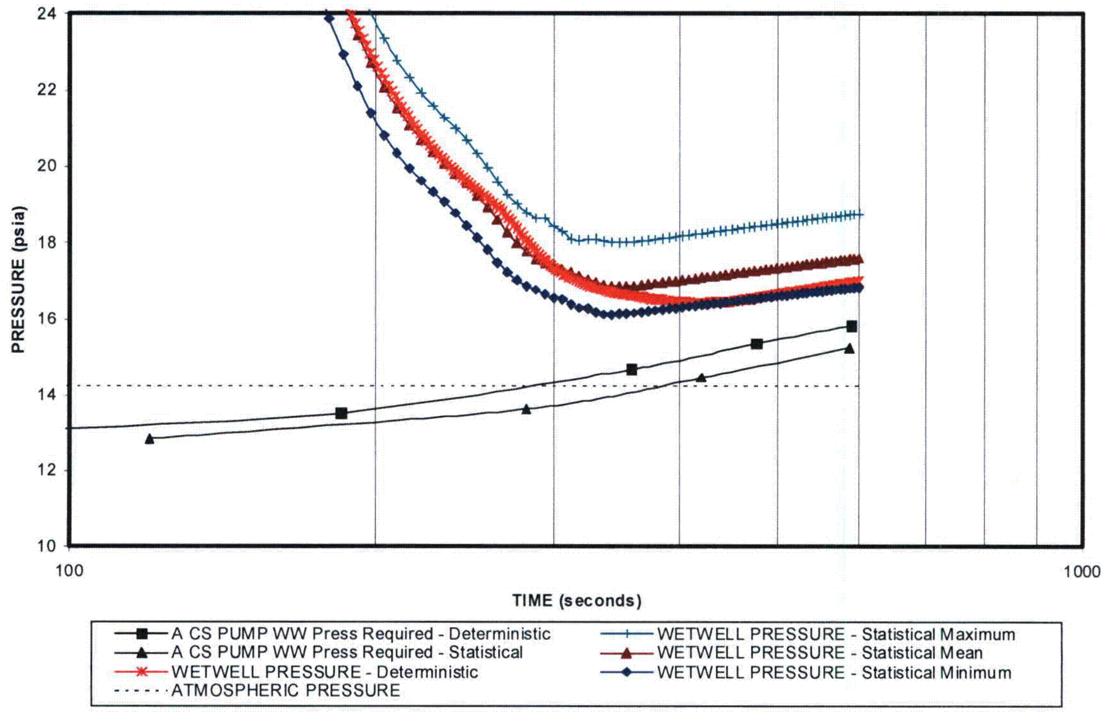

Figure A-9 Required WW Pressure vs. Available WW Pressure for CS Pump for the Short-term DBA-LOCA

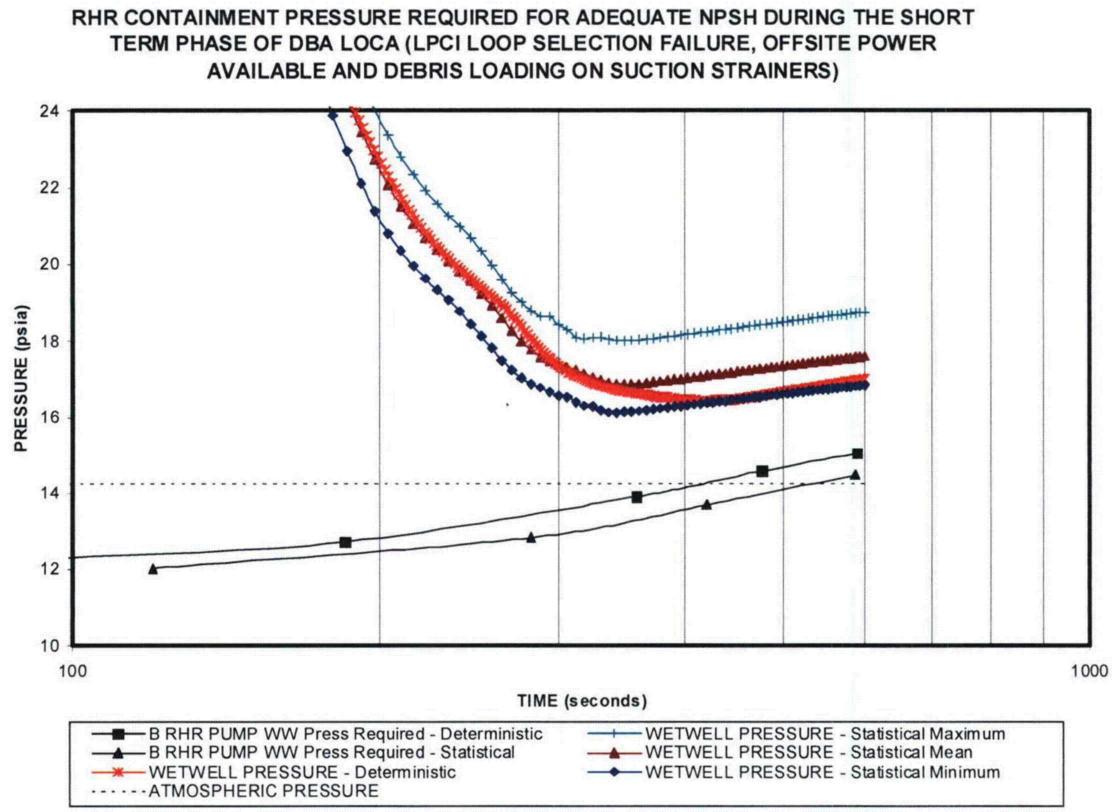

Figure A-10 Required WW Pressure vs. Available WW Pressure for LPCI Pump for the Short-term DBA-LOCA

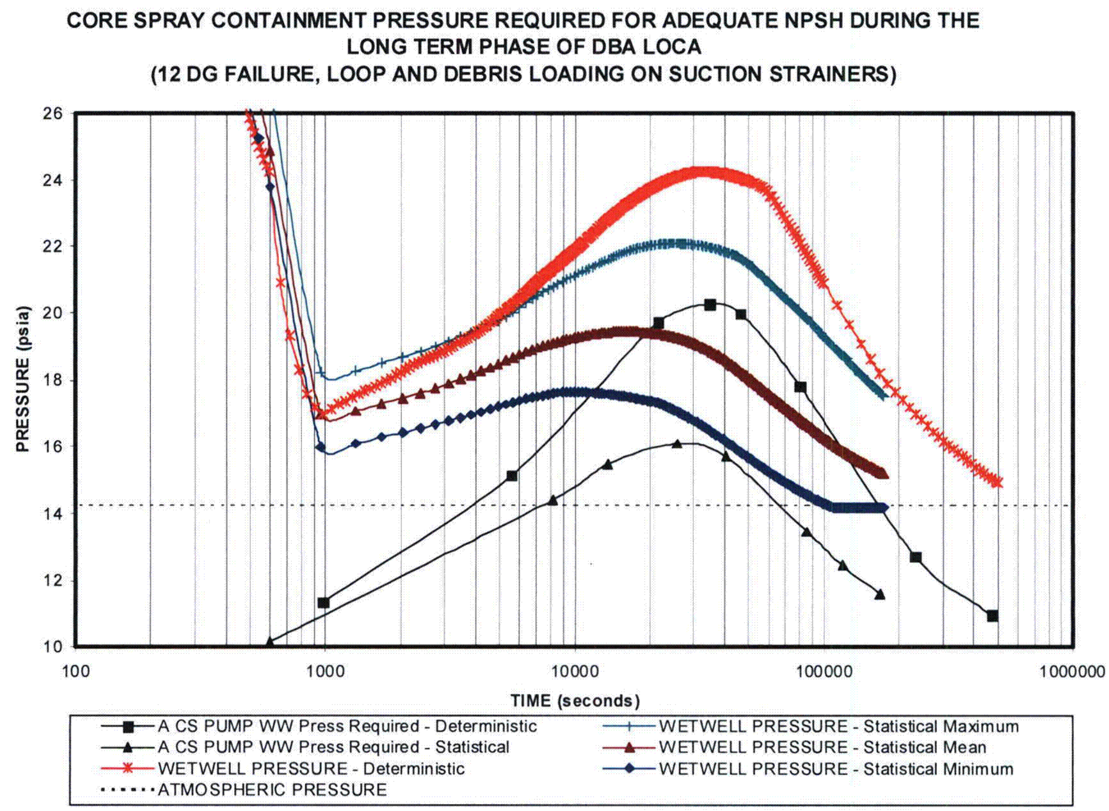

Figure A-11 Required WW Pressure vs. Available WW Pressure for CS Pump for the Long-term DBA-LOCA

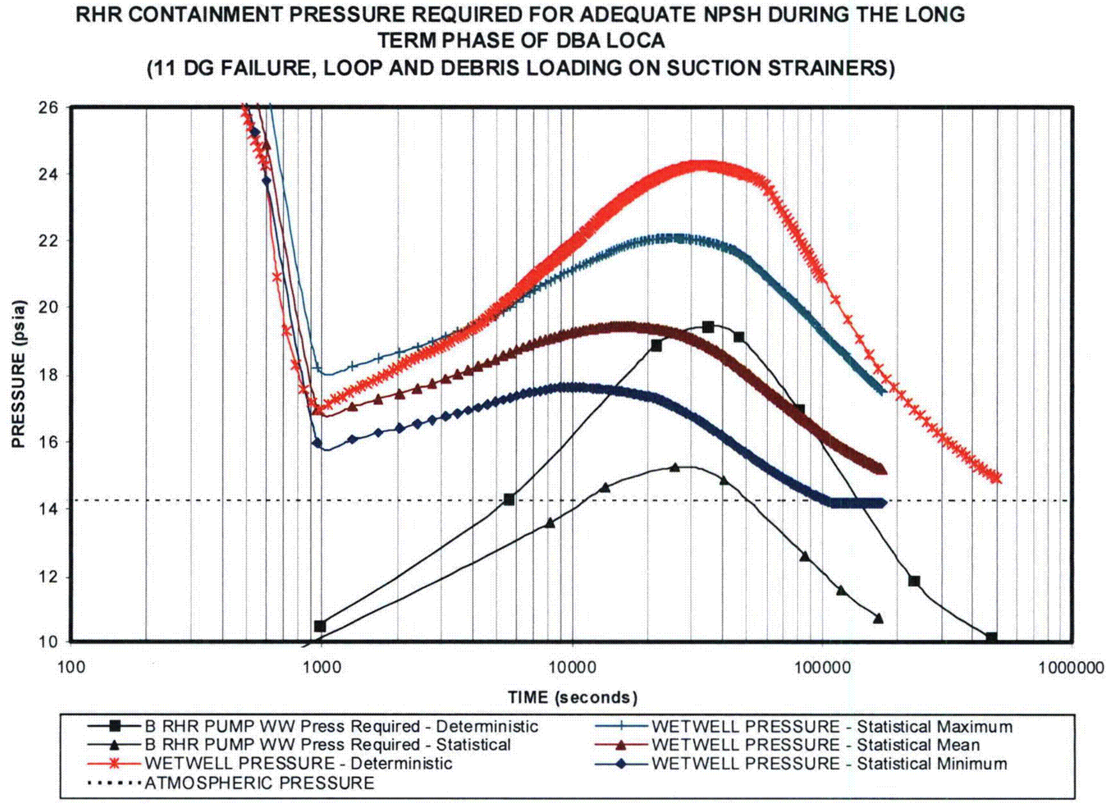

Figure A-12 Required WW Pressure vs. Available WW Pressure for LPCI Pump for the Long-term DBA-LOCA

Appendix B
DBA-LOCA Containment Response Evaluation for Use in NPSH Evaluation for Monticello Nuclear Generating Plant

It is highly improbable to have pre-existing containment leakages that are large enough to result in loss of containment overpressure, as discussed in Section 5.0. However, a concern has been expressed over whether the $NPSH_a$ would be adequate if such an improbable situation, combined with the DBA-LOCA, occurs. To address this hypothetical concern, the DBA-LOCA is analyzed, assuming no containment overpressure, but no additional failures (all safety systems are available). The intent of this evaluation is to conduct an one time demonstration. There is no intention for the Licensee to duplicate this evaluation in their licensing submittals.

Evaluation of Containment Response

The DBA-LOCA is analyzed with the following assumptions.

- A double-ended guillotine break of a recirculation line (DBA-LOCA) occurs at time zero.
- The composite analysis covering the short-term and long-term time domains is performed for the event, starting from the event inception.
- With the assumed loss of containment overpressure, the containment pressure remains near the ambient pressure throughout the event.
- All LPCI and CS pumps are available and operating.
- No LPCI flow to the broken loop occurs, as the LPCI loop selection logic functions as designed.
- At 10 minutes, the flow from four LPCI pumps is realigned in containment spray cooling with four RHR pumps, and four RHR service water pumps in service. The resulting configuration for each of the two RHR loops is one RHR heat exchanger with the flows from two RHR pumps and two RHR service water pumps.
- Runout flow for ECCS pumps occurs before 10 minutes – same as the short-term DBA-LOCA analysis described in Appendix A.
- The operator throttles the flow from ECCS pumps at 10 minutes – same as the long-term DBA-LOCA analysis described in Appendix A.

The containment response to this DBA-LOCA scenario is analyzed in two ways: the deterministic approach and the statistical approach. The analysis method is the same as the analysis presented in Appendix A. The input assumptions also are the same, except for those affected by differences in the accident scenario. Note that for this case no separate analysis is necessary between the short-term and long-term phases of the DBA-LOCA, since all safety systems including the loop selection logic are assumed to be available throughout the event.

Figure B-1 shows the value of H_{ww} with no containment overpressure (i.e., 14.26 psia above the suppression pool) as a function of time, as obtained from the deterministic approach, and from the statistical approach. At the end of this appendix, the value of $NPSH_a$ is calculated by adding the value of H_{pl} to H_{ww}, and the $NPSH_a$ calculated as such is compared with the $NPSH_r$. The suppression pool temperature response for this case is shown in Figure B-2.

B.1 Evaluation of $NPSH_a$

Section B.1 provides the value of H_{ww} (see Section 2.2 for a definition of this parameter) from the containment response evaluations based on the deterministic and statistical method, assuming that the wetwell pressure is the same as the minimum ambient pressure. The $NPSH_a$ is calculated as a function of time by adding H_{pl} (see Equation 2 in Section 2.2) to H_{ww}, and is compared with the $NPSH_r$. The value of H_{pl} is calculated according the steps described in Appendix A.

Tables B-1 and B-2 show the worst-case CS pump and LPCI pump results for the DBA-LOCA, respectively. (As mentioned before, no separate plots are necessary between the short-term and long-term DBA-LOCA.) It is noted that the available wetwell pressure is constant at the minimum ambient pressure. The results of $NPSH_a$ for the worst-case pump are also compared with $NPSH_r$ in Figures B-3 and B-4 for the CS and Figures B-5 and B-6 for the LPCI pump, respectively. The wetwell pressure results are represented by Figures B-7 and B-8 for the CS pump, Figures B-9 and B-10 for the LPCI pump and Figures B-11 and B-12 for the Limiting ECCS pump respectively.

The NPSH evaluation results for the DBA-LOCA presented in this appendix show that:

1. Based on the deterministic approach, the $NPSH_a$ is less than the $NPSH_r$ for a significant period of time.
2. Based on the statistical approach, the $NPSH_r$ for the LPCI/RHR pump stays below the $NPSH_a$ throughout the event.
3. The $NPSH_r$ for the CS pump exceeds the $NPSH_a$ only during a 4-minute period just before the assumed operator action time of 10 minutes into the event. However, the operator could throttle the CS pump flow before 10 minutes to mitigate the NPSH concern.

B.2 Conclusion

Considering that the $NPSH_r$ exceeds the $NPSH_a$ for only for a short period of time and the operator could start to throttle the CS pump flow before 10 minutes, it may be concluded that the NPSH for ECCS /RHR pumps is not of concern in a highly improbable event that a complete loss of containment overpressure occurs, combined with the DBA-LOCA.

Table B-1
Results of CS Pump NPSH Evaluation for the DBA-LOCA with Loss of Containment Overpressure

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
Short term									
110		29.75	26.6		3.15		-1.34		0.00
119	28.67		26.6	2.07		-0.88		0.00	
174	28.29		26.6	1.69		-0.72		0.00	
210		28.86	26.6		2.26		-0.96		0.00
322	25.96		26.6	-0.64		0.27		0.00	
330		27.06	26.6		0.46		-0.20		0.00
440		25.30	26.6		-1.30		0.55		0.00
454	23.87		26.6	-2.73		1.16		0.00	
570		23.92	26.6		-2.68		1.14		0.00
582	22.45		26.6	-4.15		1.76		0.00	
Long term									
660		28.57	23.0		5.57		-2.36		0.00
917	26.14		23.0	3.14		-1.33		0.00	
1380		26.62	23.0		3.62		-1.53		0.00
2147	23.71		23.0	0.71		-0.30		0.00	
2820		25.49	23.0		2.49		-1.06		0.00
5347	22.16		23.0	-0.84		0.36		0.00	
5460		25.25	23.0		2.25		-0.95		0.00
8296	22.41		23.0	-0.59		0.25		0.00	
14310		27.37	23.0		4.37		-1.86		0.00

Table B-1
Results of CS Pump NPSH Evaluation for the DBA-LOCA with Loss of Containment Overpressure

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
18660		29.00	23.0		6.00		-2.55		0.00
21660		29.73	23.0		6.73		-2.87		0.00
24060		30.03	23.0		7.03		-3.00		0.00
25673	26.02		23.0	3.02		-1.28		0.00	
47534	29.18		23.0	6.18		-2.63		0.00	

Table B-2
Results of LPCI/RHR Pump NPSH Evaluation for the DBA-LOCA with Loss of Containment Overpressure

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
Short term									
110		30.54	25.5		5.04		-2.15		0.00
119	29.46		25.5	3.96		-1.69		0.00	
174	29.07		25.5	3.57		-1.52		0.00	
210		29.64	25.5		4.14		-1.77		0.00
322	26.72		25.5	1.22		-0.52		0.00	
330		27.82	25.5		2.32		-0.99		0.00
440		26.03	25.5		0.53		-0.22		0.00
454	24.60		25.5	-0.90		0.38		0.00	
570		24.63	25.5		-0.87		-0.37		0.00
582	23.17		25.5	-2.33		0.99		0.00	
Long term									
660		26.07	22.0		4.07		-1.72		0.00
917	23.63		22.0	1.63		-0.69		0.00	
1380		24.10	22.0		2.10		-0.89		0.00
2147	21.19		22.0	-0.81		0.34		0.00	
2820		22.97	22.0		0.97		-0.41		0.00
5347	19.63		22.0	-2.37		1.00		0.00	
5460		22.72	22.0		0.72		-0.31		0.00
8296	19.88		22.0	-2.12		0.90		0.00	
14310		24.84	22.0		2.84		-1.21		0.00

Table B-2
Results of LPCI/RHR Pump NPSH Evaluation for the DBA-LOCA with Loss of Containment Overpressure

Time (sec)	$NPSH_a$ (ft)		$NPSH_r$ (ft)	NPSH Margin (ft)		Required WW Pressure (psig)		Available WW Pressure (psig)	
	Deterministic	Statistical		Deterministic	Statistical	Deterministic	Statistical	Deterministic	Statistical
18660		26.47	22.0		4.47		-1.90		0.00
21660		27.20	22.0		5.20		-2.22		0.00
24060		27.51	22.0		5.51		-2.35		0.00
25673	23.48		22.0	1.48		-0.63		0.00	
47534	26.65		22.0	4.65		-1.98		0.00	

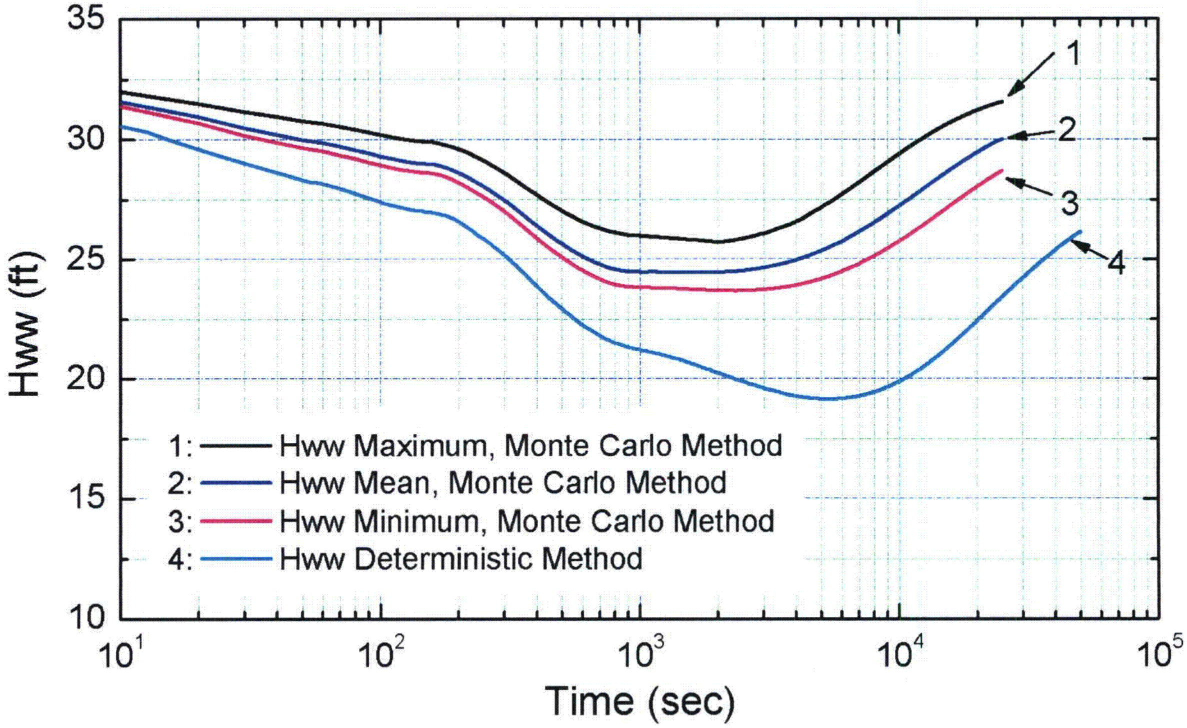

Figure B-1 Value of NPSH H_{ww} with All Safety Systems Available for Case of No Containment Overpressure

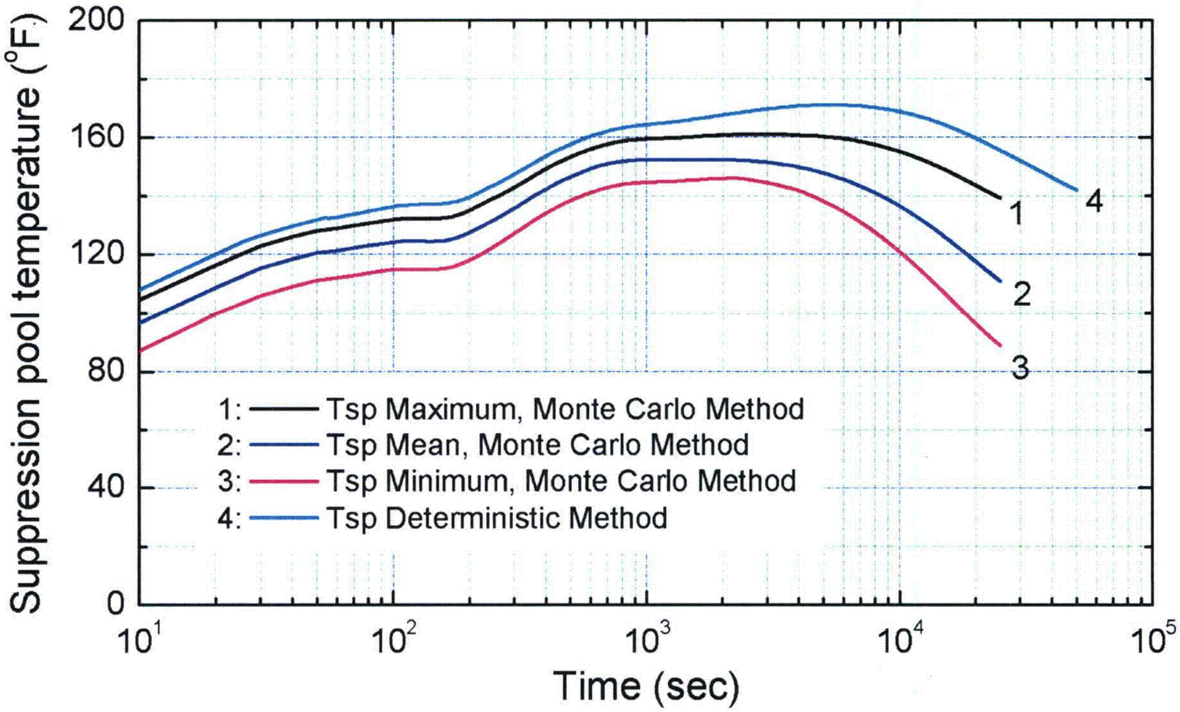

Figure B-2 Suppression Pool Temperature Response to DBA-LOCA with All Safety Systems Available for Case of No Containment Overpressure

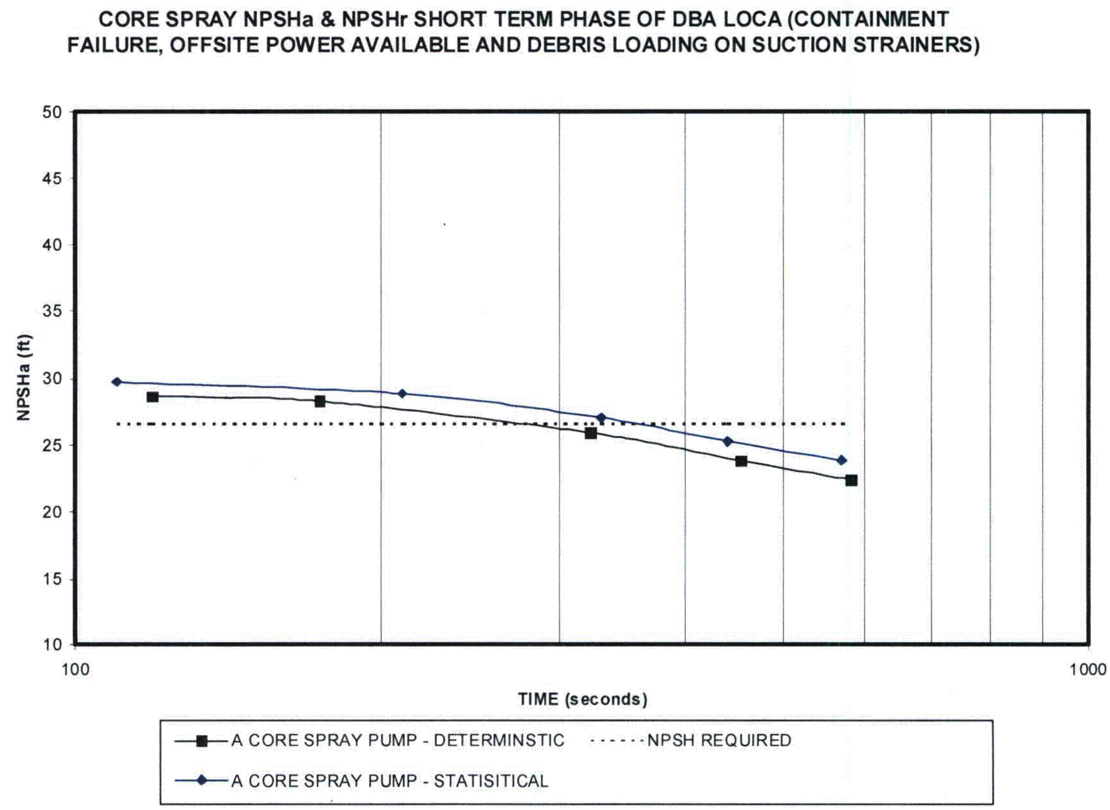

Figure B-3 Short Term CS Pump $NPSH_a$ vs. $NPSH_r$ for the DBA-LOCA with Loss of Containment Overpressure

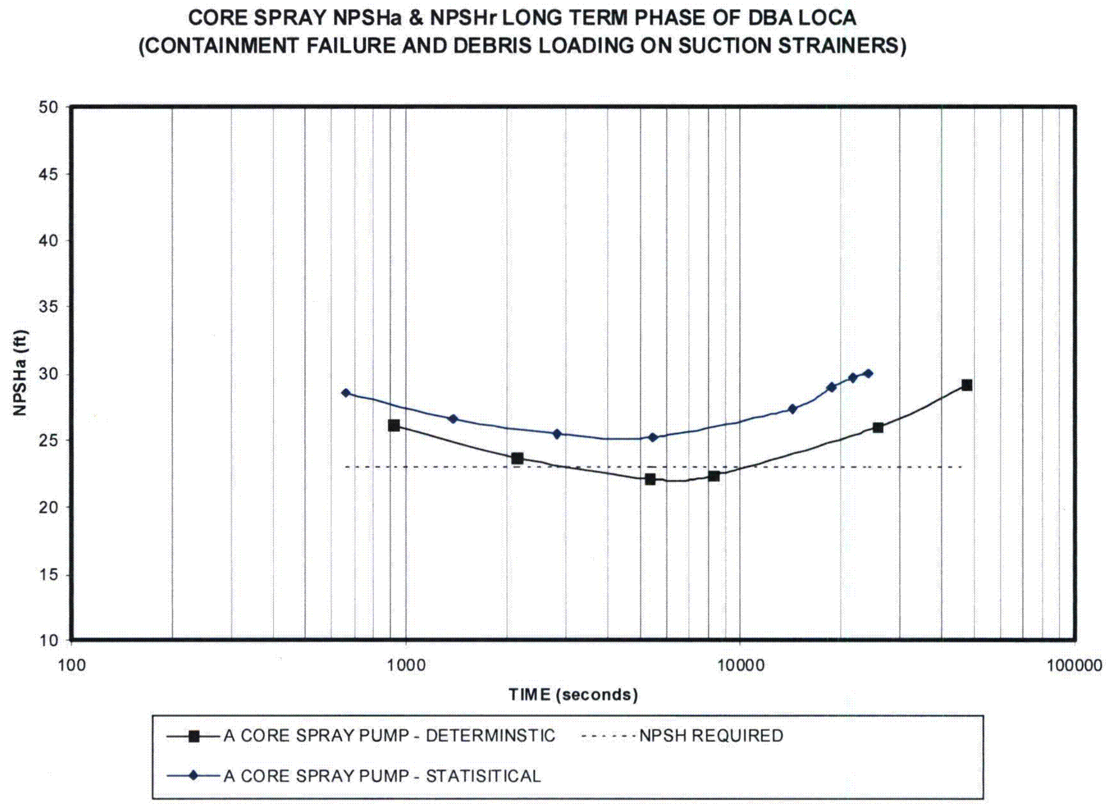

Figure B-4 Long Term CS Pump $NPSH_a$ vs. $NPSH_r$ for the DBA-LOCA with Loss of Containment Overpressure

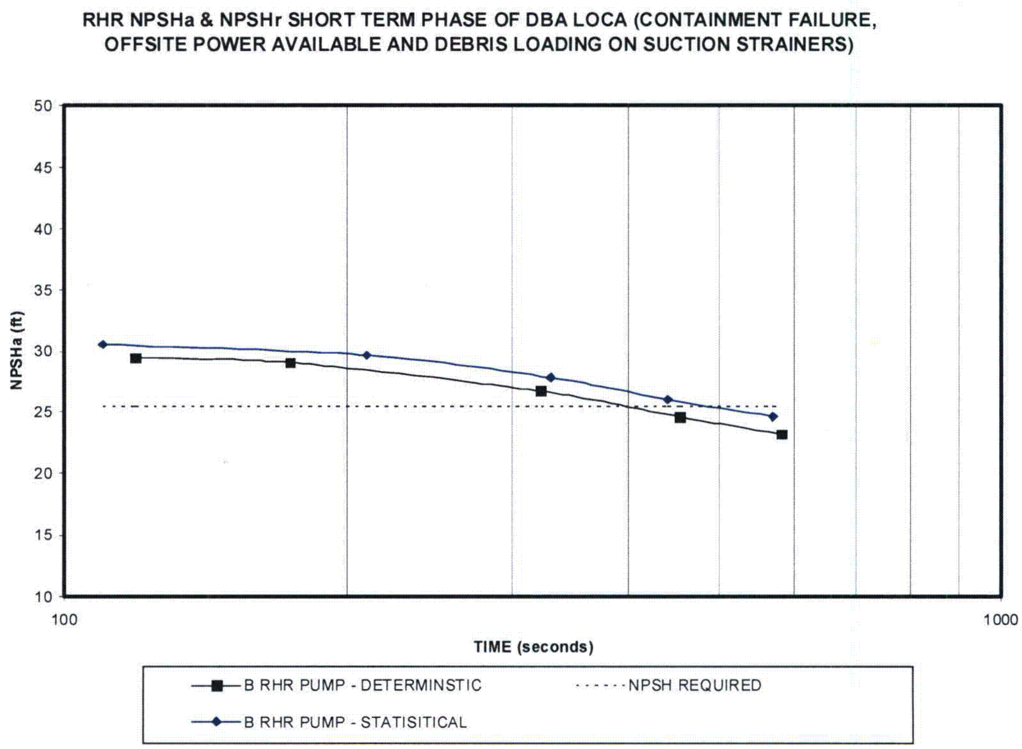

Figure B-5 Short Term LPCI Pump $NPSH_a$ vs. $NPSH_r$ for the DBA-LOCA with Loss of Containment Overpressure

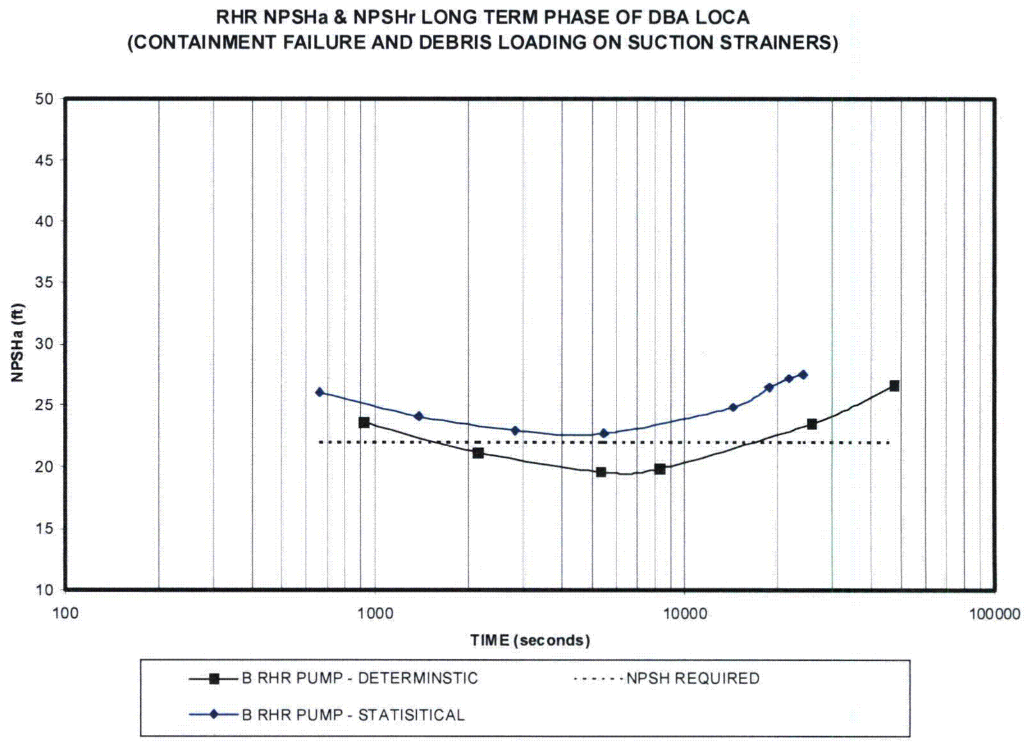

Figure B-6 Long Term LPCI Pump $NPSH_a$ vs. $NPSH_r$ for the DBA-LOCA with Loss of Containment Overpressure

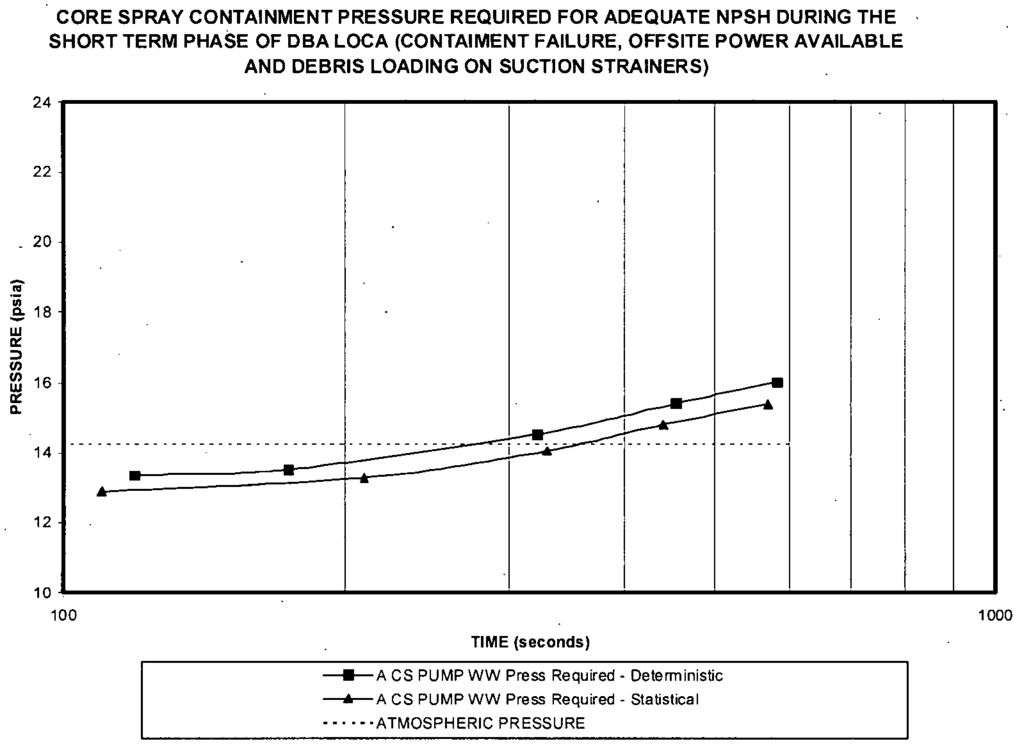

Figure B-7 Short Term Required WW Pressure vs. Available WW Pressure for CS Pump during the DBA-LOCA with Loss of Containment Overpressure

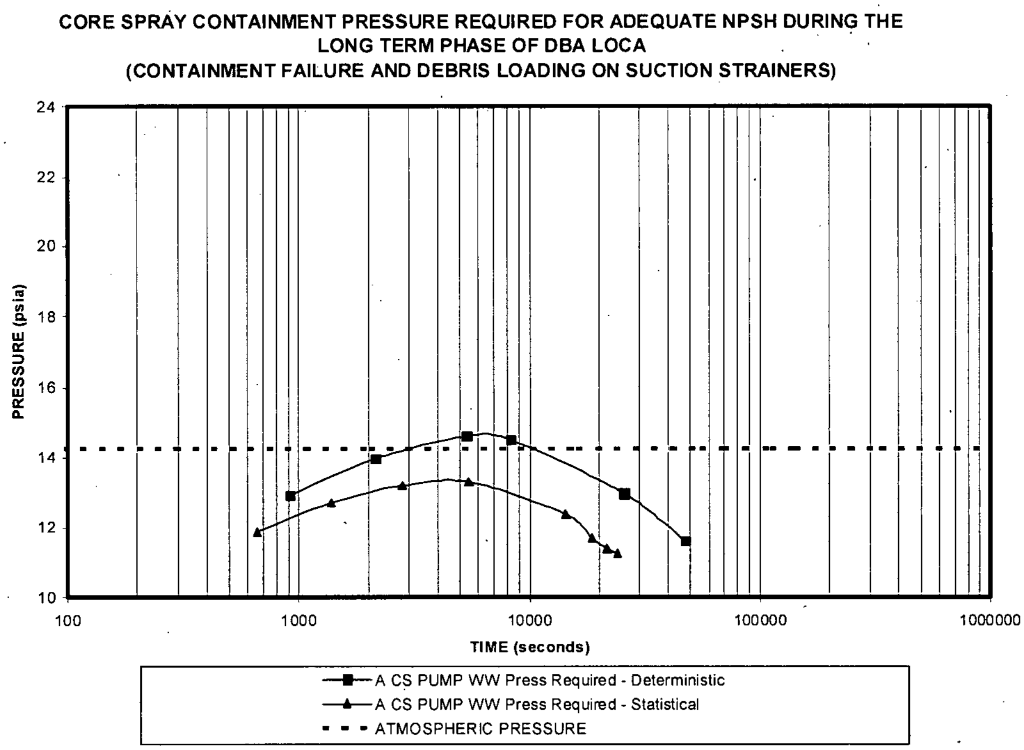

Figure B-8 Long Term Required WW Pressure vs. Available WW Pressure for CS Pump during the DBA-LOCA with Loss of Containment Overpressure

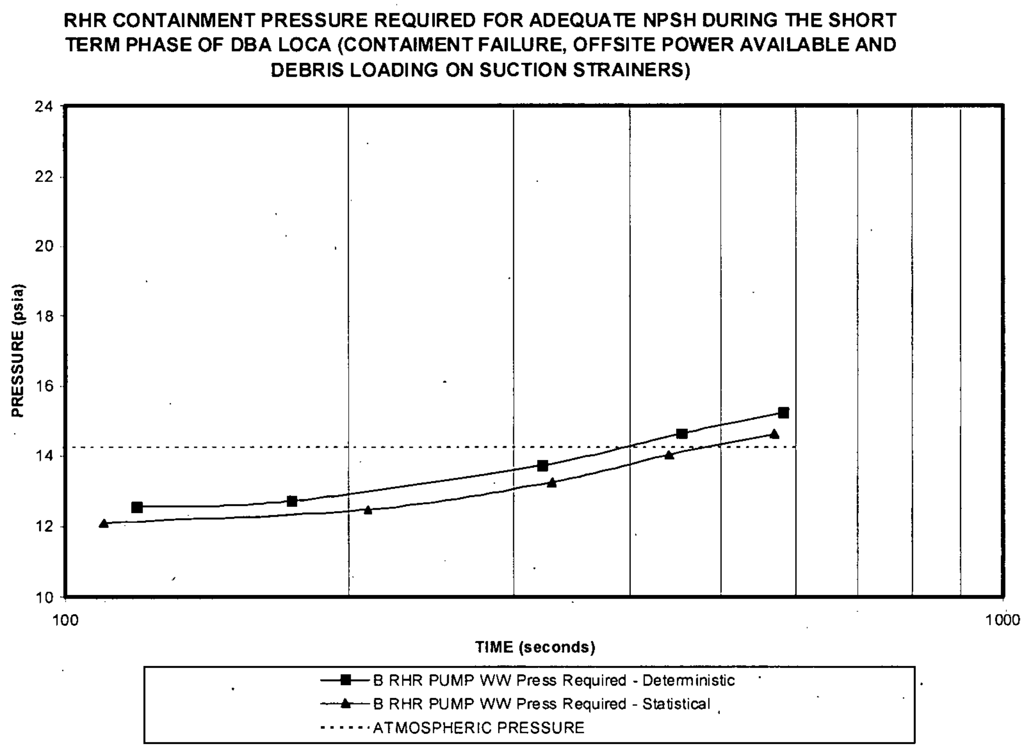

Figure B-9 Short Term Required WW Pressure vs. Available WW Pressure for LPCI Pump during the DBA-LOCA with Loss of Containment Overpressure

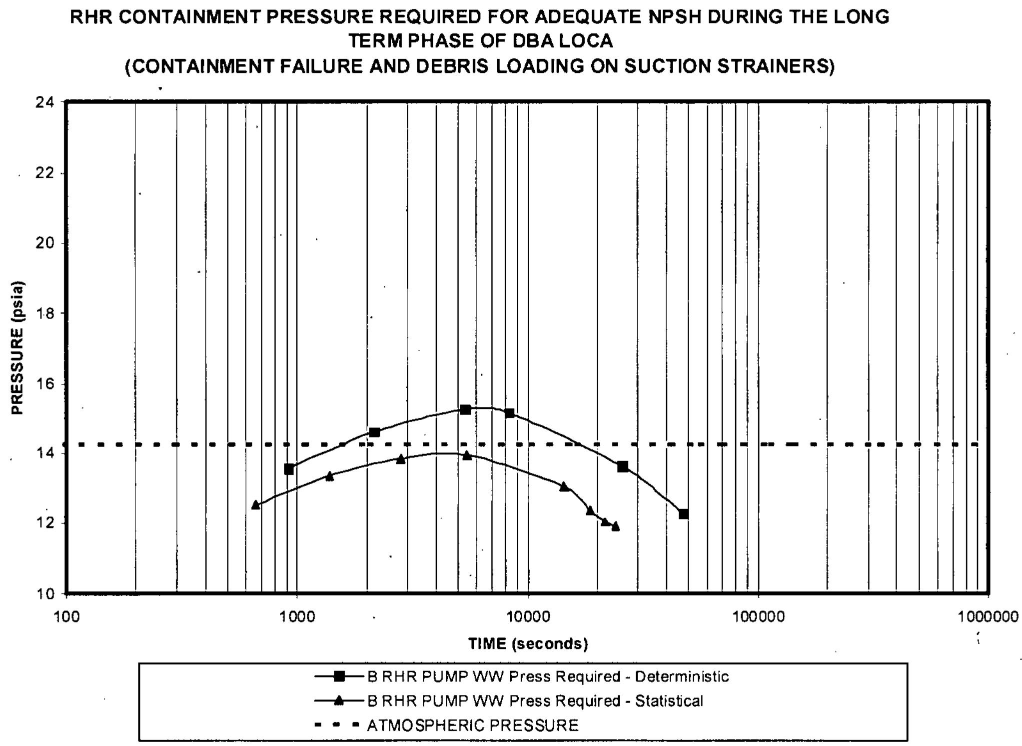

Figure B-10 Long Term Required WW Pressure vs. Available WW Pressure for LPCI Pump during the DBA-LOCA with Loss of Containment Overpressure

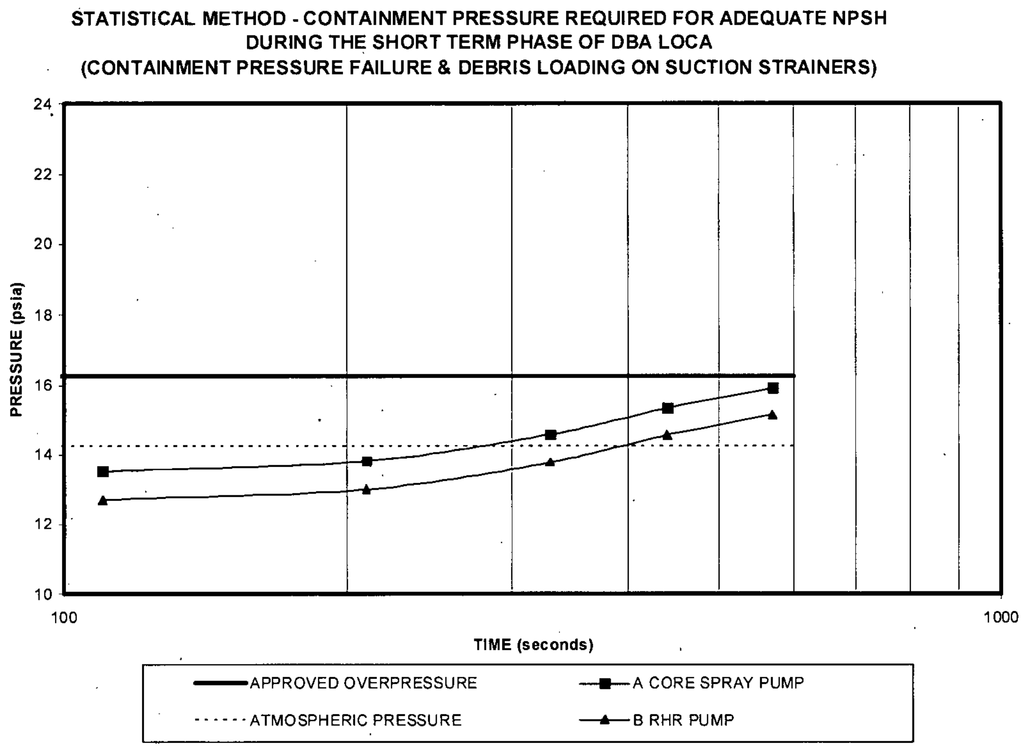

Figure B-11 Short Term Required WW Pressure vs. Available WW Pressure for Limiting ECCS Pumps during the DBA-LOCA with Loss of Containment Overpressure

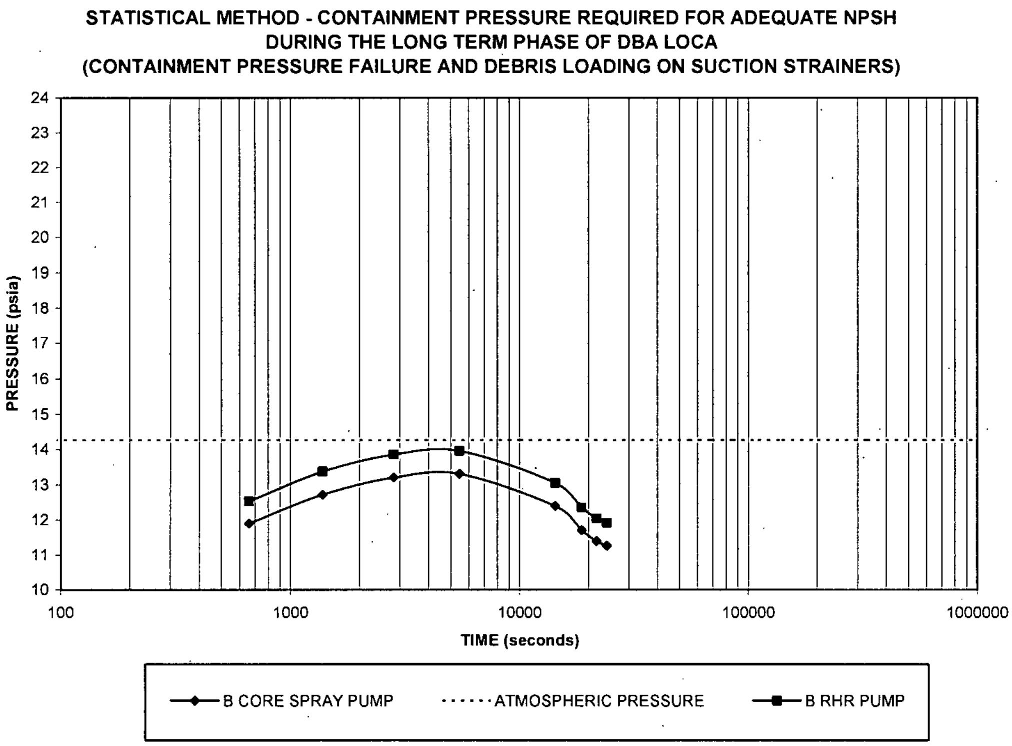

Figure B-12 Long Term Required WW Pressure vs. Available WW Pressure for Limiting ECCS Pumps during the DBA-LOCA with Loss of Containment Overpressure

APPENDIX C Alternate Methods To Containment Overpressure CreditAlternate Methods to Containment Overpressure Credit

Operator actions are available that have the potential to either enhance available post-accident containment overpressure or alternatively increase available NPSH and therefore reduce or eliminate ECCS pump reliance on containment overpressure to maintain adequate NPSH. This Appendix describes both types of potential actions.

C.1.1 Methods That Enhance Containment Overpressure

In general, these methods should not be considered for design basis events, they could be considered for special events and beyond design basis events. $NPSH_a$ can be increased by one or more of the following methods:

1. [[

2.

3.

]]

C.1.1.1 Introduction of Additional Non-Condensable Gases

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C.1.1.2 Reducing Containment Cooling to Increase Containment Heat Input

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C.1.1.3 Suppression Chamber Water Addition to Improve $NPSH_a$

[[

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Appendix D
Containment Overpressure Credit

D.1 Methods to Consider for Reducing or Eliminating Need for Containment Overpressure Credit

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D.2 Improved Containment (Torus) Cooling

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D.3 Options that Increase in $NPSH_a$ /Reduce $NPSH_r$

[[

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D.4 Qualification Testing

[[

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D.5 Pump Replacement

The ultimate alternative to COP is to replace existing ECCS pumps with a new pump design (e.g. vertical turbine) in a configuration that provides adequate NPSH under postulated post-accident containment conditions, without credit for containment overpressure.

Table D-1
Summary of Plant Data

PLANT	PUMP TESTED	TEST SUMMARY AND RESULTS
[[		

Table D-1
Summary of Plant Data

PLANT	PUMP TESTED	TEST SUMMARY AND RESULTS

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