POLICY ISSUE
(Notation Vote)

January 31, 2011 SECY-11-0014

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: USE OF CONTAINMENT ACCIDENT PRESSURE IN ANALYZING EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT HEAT REMOVAL SYSTEM PUMP PERFORMANCE IN POSTULATED ACCIDENTS

PURPOSE:

The purposes of this paper are to: (i) respond to the staff requirements memorandum (SRM) concerning the U.S. Nuclear Regulatory Commission's (NRC's) meeting with the Advisory Committee on Reactor Safeguards (ACRS), dated June 25, 2010 (SRM M100609B), and (ii) request Commission approval of the staff's plan to resolve issues regarding the use of containment accident pressure (CAP) in analyzing pump performance in emergency core cooling systems (ECCSs) and containment heat removal systems during postulated accidents. These issues arise from continuing discussions between the NRC staff (staff) and ACRS.

SUMMARY:

The staff has developed options for addressing the issues on CAP. The staff recommends the first option, which is to resume review of the two license amendment requests currently on hold because of the issues on CAP, and to use its existing risk review guidance, including the review

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guidance for nonrisk-informed applications, consistent with the existing direction from the Commission and the staff's recently-developed deterministic guidance for future CAP reviews.

In SRMs regarding two meetings with ACRS, on June 5, 2008 (SRM M080605B), and November 7, 2008 (SRM M081107), the Commission directed the staff to continue working to resolve differences between ACRS and the staff concerning the use of CAP in demonstrating that pumps in ECCSs and containment heat removal systems can perform their safety function(s). In the most recent SRM on this subject (SRM M100609B), the Commission directed the staff to discuss the areas in which it "aligns with the ACRS and disagrees with the ACRS" with regard to the use of CAP, including defense-in-depth implications, the use of risk information, and whether to assess the practicability of plant modifications to eliminate the need for CAP credit.

The remainder of this paper discusses these areas and the options for addressing the overall CAP issue. Enclosure 1 provides a regulatory history of the CAP issue and technical and regulatory information on the use of CAP in reactor safety analyses. It forms the basis upon which the staff is ready to proceed with the recommended approach. Enclosure 2 discusses areas of agreement and disagreement between the staff and ACRS other than the three issues specifically identified in SRM M100609B. The staff addresses those three issues in this paper.

BACKGROUND:

The accident analyses for many operating reactors rely on pressure higher than that present before the postulated accident to provide net positive suction head (NPSH) margin for the pumps in the ECCS and the containment heat removal system. NPSH margin is a measure of the pumps ability to avoid excessive cavitation so that it can perform its safety function(s). Section 2.0 of Enclosure 1 to this paper describes NPSH and cavitation in more detail.

In calculating NPSH margin, the inclusion of some or all of the pressure developed in the containment during an accident is referred to as CAP credit. The staff earlier used the term "containment overpressure" for this issue but discontinued its use for two reasons: (1) industry uses several definitions of containment overpressure, and (2) the term has been confused with exceeding the design pressure of the containment. The containment design pressure is never exceeded while crediting CAP.

The total containment accident pressure is made up of two partial pressures: the partial pressure of the water vapor and the partial pressure of the heated air or nitrogen present in the containment atmosphere before the postulated accident. For pressurized water reactors (PWRs), the vapor pressure during a loss-of-coolant accident (LOCA) (determined by the sump water temperature) is predicted to be greater than the total containment pressure before the postulated accident. This vapor pressure is credited to demonstrate adequate NPSH margin. Some PWRs also credit a portion of the air partial pressure to ensure adequate NPSH margin. For boiling water reactors (BWRs) both the partial pressure of the vapor in the containment atmosphere and the air partial pressure are used to demonstrate adequate NPSH margin for the LOCA and certain "special events" (station blackout, anticipated transients without scram, and safe shutdown following fires). The amount of CAP credited and its duration depend on pump and system characteristics. These vary from plant to plant.

Overview of Regulatory History

Some early reactor designs credited CAP in calculating NPSH margin. In response to an ACRS recommendation to discontinue this practice, the NRC issued Regulatory Guide (RG) 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," in November 1970 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML052440069). The position in RG 1.1 is that the containment pressure used to determine NPSH margin should be the pressure in containment before the accident. However, the NRC did not backfit this position on plants that were already licensed with credit for CAP, and the staff allowed credit for CAP throughout the years because of various operating reactor conditions and in response to specific technical issues. (See Section 4.2 of Enclosure 1).

The guidance in RG 1.82, "Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident," Revision 3, issued November 2003 (ADAMS Accession No. ML033140347), retained the regulatory position in RG 1.1 (i.e., the ECCS and the containment heat removal system should be designed to provide sufficient NPSH margin without any increase in containment pressure from that present before the accident). However, RG 1.82, Revision 3, also recognizes that some operating nuclear power plants for which modifications to avoid crediting CAP are impracticable, may need CAP credit to demonstrate adequate NPSH margin.

Extent of CAP Credit

The table below provides the current status of the use of CAP for operating reactors.

Status of Use of Containment Accident Pressure for Currently-Operating Reactors		
	Pressurized Water Reactors	Boiling Water Reactors
RG 1.1 position met	9	16
RG 1.1 position not met	60	19

Interactions between the Staff and the Advisory Committee on Reactor Safeguards

Although licensees have taken credit for CAP in various license amendments submitted in recent years, ACRS and the staff have focused their discussions on license amendment requests for extended power uprates (EPUs). A power uprate increases the decay heat level following a reactor trip. This results in an increase in the temperature of the sump water (PWRs) or suppression pool water (BWRs). This reduces the NPSH margin.

The staff has discussed the use of CAP with ACRS on many occasions, both generically and as it applies to specific operating reactors. Section 4.2 of Enclosure 1 summarizes the discussions since 1997.

Enclosure 2 summarizes areas of agreement and disagreement between the staff and ACRS other than the three topics explicitly mentioned in SRM M100609B. The staff has implemented several ACRS recommendations. The ACRS letter, "Crediting Containment Overpressure in

Meeting the Net Positive Suction Head Required to Demonstrate that the Safety Systems Can Mitigate the Accidents as Designed," dated March 18, 2009, to the NRC Executive Director for Operations (EDO) (ADAMS Accession No. ML090700464) recommends that, if the amount of CAP that must be credited based on licensing analyses is not a small fraction of the total CAP and limited in duration (amount and duration both not defined), the licensee should provide additional information to address conservatisms and explicitly account for uncertainties. In consultation with two pump industry experts on NPSH and cavitation, the staff developed interim staff guidance that quantifies uncertainty and margin in using CAP. (See Section 6.0 of Enclosure 1). The staff discussed this interim staff guidance with ACRS at the 572nd ACRS meeting on May 6, 2010. The staff also transmitted this guidance to the BWR Owners' Group (BWROG) (ADAMS Accession No. ML100550903) and the PWR Owners' Group (ADAMS Accession No. ML100740516) for their comment. ACRS stated, in the May 19, 2010, letter, "Draft Guidance on Crediting Containment Accident Pressure in Meeting the Net Positive Suction Head Required to Demonstrate that Safety Systems Can Mitigate Accidents as Designed," to the NRC EDO (ADAMS Accession No. ML101300332), that this guidance "provides an improved framework for a more comprehensive assessment of crediting containment accident pressure in meeting NPSH requirements."

At the staff's request, BWROG developed a statistical method for analyzing the use of CAP for BWRs that also quantifies the NPSH margin. A BWROG topical report (ADAMS Accession No. ML080520261) describes this method. The staff and BWROG discussed this report with ACRS at the 572nd ACRS meeting. BWROG may revise this topical report to include the recently-developed deterministic staff guidance.

The staff also performed a generic risk analysis of a BWR with a Mark I containment. The results show that the risk increase caused by using CAP is very small. (See Section 5.5 of Enclosure 1 and the transcript of the May 6, 2010, ACRS meeting). The May 19, 2010, ACRS letter stated that "the PRA [probabilistic risk assessment] studies by the staff are helpful in assessing the importance of pre-initiator and post-initiator leak probability and leakage test interval on the changes in risk associated with CAP credit." ACRS noted that the staff analysis did not include postulated accidents initiated by seismic events, operator actions that could affect CAP, or fires.

DISCUSSION:

The Commission requested a discussion of three issues related to the use of CAP for which the staff and ACRS are not in agreement. Issue 1 addresses defense-in-depth as it relates to this issue because it represents the fundamental consideration in the use of CAP. Issue 2 concerns the use of risk information to judge the acceptability of CAP credit. Finally, Issue 3 discusses whether the staff should consider the practicability of hardware modifications to eliminate the need for CAP credit.

Issue 1: Application of CAP with Respect to Defense-in-Depth

The staff understands the ACRS position on defense-in-depth, which is to maintain the independence of the containment function and the accident mitigation function by not relying on containment accident pressure (transcript of NRC Commission meeting with the ACRS, June 9, 2010 (ADAMS Accession No. ML101660107), p. 31).

Defense-in-depth is a basic element of the NRC's safety philosophy. Defense-in-depth has been applied in various forms. One application of defense-in-depth is to ensure that key safety functions do not depend on a single element of design, construction or operation. Another form of the defense-in-depth philosophy is a balance among accident prevention, accident mitigation and the limitation of the consequences of an accident. Redundant and diverse means may be used to accomplish key safety functions. One manifestation of defense-in-depth is the use of multiple independent fission product barriers.

Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued July 1998, provides guidance on the evaluation of defense-in-depth attributes in the review of risk-informed license amendment requests. One of these attributes is that "the independence of barriers is not degraded." However, the regulations do not specify that the fission product barriers be independent.

The use of CAP credit for ECCS and containment heat removal pump NPSH margin has the potential to create a dependency between the containment and another fission product barrier (the fuel cladding) because containment leakage, sufficiently greater than that allowed by technical specifications, could result in insufficient CAP and failure of the ECCS and containment heat removal pumps to perform their safety functions because of excessive cavitation. This pump failure could, in turn, lead to core damage.

A dependency between fission product barriers created by the use of CAP credit may adversely impact defense-in-depth. First, the key safety function of providing abundant emergency core cooling operation depends, for many plant designs, on an intact containment as a source of water (e.g., suppression pools for BWRs; containment sumps for PWRs during the recirculation phase). Failing the containment in a manner that would prevent use of this water (e.g., a failure at the bottom of a BWR suppression pool) could impair the ECCS function. The need for CAP credit could increase the level of dependency between the containment and the ECCS. This would be a reduction in defense-in-depth because a single element of the design (i.e., the containment) is relied upon to ensure this key safety function.

An increase in dependence between the containment and the ECCS could also affect the balance among accident prevention, accident mitigation, and limiting accident consequences. Crediting CAP would have no impact on accident initiation, but containment failure could impair the ability of the ECCS to mitigate the accident. Accident consequences as determined by design-basis calculations could also be adversely impacted by CAP credit, because the failure of containment could not only result in fuel failure but also reduction in the effectiveness of the final fission product barrier.

Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," reflects the defense-in-depth principles, although Appendix A does not explicitly refer to defense-in-depth. A balance among accident prevention, accident mitigation, and limiting accident consequences is basic to the general design criteria. Specific requirements in the general design criteria exist for independence, redundancy, and diversity (oftentimes achieved by imposing the requirement to withstand a "single failure)." The general design criteria also require a level of quality commensurate with the safety functions of structures, systems, and components and require the capability for inspection and testing. These requirements ensure that the individual fission product barriers are capable of performing their safety functions. However, the general design criteria do not explicitly require the independence of fission product barriers, and CAP credit does not contravene any general design criterion.

The regulation at 10 CFR 50.48, "Fire protection," characterizes defense-in-depth for fire protection, and 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors," mentions defense-in-depth but does not define it.

The International Atomic Energy Agency publication INSAG-10, "Defense in Depth in Nuclear Safety," (issued 1996), describes the implementation of defense-in-depth and illustrates how its various elements interrelate. INSAG-10 implements the defense-in-depth philosophy consistent with 10 CFR Part 50 of the NRC's regulations. INSAG-10 also does not specify the independence of fission product barriers. Instead, it states that, for situations in which the fission product barriers are not fully independent, their reliability should be greater.
As discussed in Section 5.1 of Enclosure 1, great care is taken to ensure the integrity (leak tightness) of the containment in design, construction and operation. For this reason, containment integrity is assumed in all design-basis accident analyses. The consideration of containment failure implies a beyond-design-basis accident. Several instances of containment structure degradation have occurred over the past 25 years, and a number of NRC generic communications and reports have addressed them. Most of the recent events have involved localized corrosion of the carbon steel liner of concrete containments, some with (liner) through-wall penetration, although with very small measured or calculated leakage. None of the events involved a loss of containment design function, including leak tightness assumed in the dose analyses. The NRC staff continues to monitor the industry response to these events, especially with regard to inspection methods and frequency.

In fact, several regulations assume the integrity of the containment to satisfy important safety limits. For example, Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 credits containment pressure in LOCA calculations. This assumes containment integrity. The offsite dose limits in 10 CFR Part 100, "Reactor Site Criteria," or 10 CFR 50.67, "Accident source term," might not be met following a LOCA if the containment leakage rate is greater than the very small leakage rate specified in Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. This small leakage rate implies containment integrity.

Section 4.1 of Enclosure 1 discusses defense-in-depth in NRC regulations and the application of defense-in-depth to reactor safety in other documents.

Issue 2: Use of Risk Information to Assess Appropriateness of CAP Credit

The staff understands the ACRS position on risk to be that, if hardware modifications are impractical to eliminate the need for CAP, then the defense-in-depth margins that are involved in allowing CAP credit should be relaxed only if the associated increase in risk is small. The risk assessment should be plant-specific, should consider the risk from fire and seismic as well as internal events, and should consider the possibility of operator errors that could affect CAP (transcript of NRC Commission meeting with the ACRS, June 9, 2010 (ADAMS Accession No. ML101660107), pp. 32, 34).

The Issue 1 discussion states that the use of CAP credit potentially affects defense-in-depth by increasing the level of dependence between the fuel cladding and the containment. Risk information could be used as a measure of the significance of reducing the independence of fission product barriers because of CAP in a given license amendment request. A PRA approach could be used to assess the increase in risk that could result from a failure of containment that disabled the ECCS by removing the needed CAP. A PRA is ideally suited to assess the balance among accident prevention, mitigation, and consequences. However, no NRC regulation requires licensees to provide risk information related to crediting CAP as part of the basis for supporting an amendment to a Part 50 operating license.

In fact, license amendment requests crediting CAP are typically not risk-informed (i.e., submitted in accordance with the guidance in RG 1.174). SRP Section 19.2, Appendix D, requires the NRC staff to find “special circumstances that could rebut the presumption of adequate protection” to require risk analysis for nonrisk-informed license amendment requests. The staff submitted this approach to the Commission in SECY-99-246, “Proposed Guidelines for Applying Risk-Informed Decision Making in License Amendment Reviews,” dated October 19, 1999. The Commission approved the staff recommendation and provided additional implementation guidance in its related SRM.

ACRS recommends a plant-specific PRA to justify the use of CAP for NPSH margin. In crediting CAP, the staff has not identified special circumstances that could rebut the presumption of adequate protection afforded by compliance with the regulations. This is supported, in part, by a generic risk analysis that the staff performed on a BWR with a Mark I containment. This assessment concluded that the risk of using CAP for this plant is very small. (See Section 5.5 of Enclosure 1). Therefore, the staff has no basis within the current regulations or established Commission policy to require licensees to perform a plant-specific PRA to support crediting CAP. However, if the NRC can demonstrate that there is a need, justifiable on both technical and backfitting grounds, for requesting risk information in order to make a regulatory decision on the licensee’s request to use CAP, then the NRC may request a risk analysis assessing the use of CAP to address licensee-initiated license amendments that depend upon CAP (or an extension of CAP to new applications)

Issue 3: Practicability of Hardware Changes to Eliminate CAP Credit

The staff understands the ACRS position on practicability to be that licensees should be required to demonstrate, on a plant-specific basis, that it is not practical to reduce or eliminate the need for overpressure credit by hardware changes or requalification of equipment before NRC would consider granting CAP credit (transcript of NRC Commission meeting with the ACRS, June 9, 2010 (ADAMS Accession No. ML101660107), p. 31).

RG 1.82, Revision 3, includes a regulatory position that ECCSs and containment heat removal systems should be designed to provide sufficient available NPSH to the system pumps, assuming the maximum expected temperature of pumped fluid and no increase in containment pressure from that present before the postulated LOCA. The regulatory position also states that this may not be possible for certain operating reactors for which the design cannot be practicably altered. RG 1.82, Revision 3, allows CAP credit in such cases, if supported by an acceptable technical analysis.

The issue is whether the staff should make judgments about the practicability of plant modifications to avoid using CAP credit in NPSH margin calculations.

The NRC staff reviews license amendment requests and performs a safety evaluation based on regulations, regulatory guidance, and Commission policy. The staff does not judge whether the licensee could have met its objective in some other way. The staff only evaluates the safety of the licensee's proposal. This is consistent with 10 CFR 50.109, "Backfitting." Even when the NRC imposes a backfit, the licensee is ordinarily free to choose a compliance option that best suits its purposes (10 CFR 50.109(a)(7)).

New Reactors

In SRM SECY-06-0144, "Proposed Reorganization of the Office of Nuclear Reactor Regulation and Region II," dated July 21, 2006, the Commission directed the staff to consistently apply technical and regulatory standards, guides, and requirements both to new plant licensing and to operating plants and to look for other strategies, as appropriate, to achieve and maintain the desired consistency.

The only new reactor designs that have credited CAP to demonstrate adequate NPSH margin are the active PWR designs. Passive plants do not rely on active pumping systems, and the active BWR has not taken credit for containment pressure greater than the pressure before the postulated accident in NPSH analyses.

Active PWR designs for new reactors take a similar approach to the use of CAP in evaluating NPSH as operating PWR designs do. For example, both operating PWRs and new active PWR designs predict elevated sump water temperatures (greater than 100 degrees Celsius (212 degrees Fahrenheit)) and elevated pressures in containment during a design-basis accident. In calculating NPSH margin to demonstrate it is sufficient, new active PWR designs, similar to most operating PWRs, credit the inclusion of vapor pressure developed in containment during an accident in the NPSH margin calculation. (See Section 2.0 of Enclosure 1).

The regulation at 10 CFR 52.47, "Contents for applications; technical information," effectively requires design certification applicants to perform PRAs; an analogous regulatory requirement in 10 CFR 52.79 exists for combined license applicants. Therefore, the staff can request both design certification and combined license applicants to assess the risk significance of using CAP using the required PRA.

The staff will review the use of CAP in NPSH evaluations in the same manner for new reactors and operating plants, except for the treatment of risk.

CAP Regulatory Guidance

The staff's guidance on CAP appears in several NRC documents. Guidance to external stakeholders is provided in RG 1.1 and RG 1.82, Revision 3. CAP guidance to the NRC staff appears in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 6.2.2, "Containment Heat Removal Systems," Revision 5, issued March 2007, and in the Office of Nuclear Reactor Regulation's document, RS-001, Revision 0, "Review Standard for Extended Power Uprates," dated December 24, 2003 (ADAMS Accession No. ML033640024). Based on the Commission's decision on this issue and after further discussions with ACRS, the staff intends to revise both the regulatory guides and the SRP to ensure consistency. For example, RG 1.1 states that only the pressure before a postulated accident should be used, while RG 1.82 Revision 3 permits an exception for operating reactors for which plant modifications to avoid using CAP are impracticable. Also, SRP Section 6.2.2 provides acceptance criteria for containment spray pumps if CAP is credited in NPSH evaluations, but it does not mention the ECCS pumps. In addition, SRP Section 6.2.2 mentions performing an evaluation of the contribution to plant risk for CAP, which is not consistent with SRP Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance,” Appendix D, “Use of Risk Information in Review of Non-Risk-Informed License Amendment Requests,” issued June 2007, and the Commission’s SRM in SECY-99-246, “Proposed Guidelines for Applying Risk-Informed Decision Making in License Amendment Reviews,” dated January 5, 2000, and which is not mentioned in other guidance documents. (See the Issue 2 discussion).

Backfitting Considerations

The regulation at 10 CFR 50.109 requires a backfit analysis of any proposed change or new requirement except in three cases: (1) the backfitting is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee, (2) backfitting is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security, or (3) the backfitting involves defining or redefining what level of protection to the public health and safety or the common defense and security should be regarded as adequate. The CAP credit issue does not rise to the level of questioning adequate protection, and the use of CAP does not constitute noncompliance with applicable NRC requirements. Therefore, to impose a backfit on licensees currently using CAP, the NRC must demonstrate under § 50.109(a)(3) that: (1) imposition of new NRC requirements constitutes a substantial increase in the overall protection of the public health and safety or the common defense and security, and (2) the direct and indirect costs of implementing the regulatory action are justified in view of this increased protection.

Options to Address the CAP Issue

The staff has identified the following two options for addressing the CAP issue for operating reactor reviews. New reactor reviews will continue and deviations from guidance will be reviewed on a case by case basis.

(1) The staff resumes work on EPU applications. The staff’s evaluation of current EPU applications, as well as future applications for new or increased credit for CAP, would be consistent with staff practice in implementing the current risk review guidance (SRP Section 19.2), including the review of nonrisk-informed applications such as EPUs (Appendix D of SRP Section 19.2) and the recently-developed deterministic guidance based on ACRS recommendations to include uncertainty and margins in CAP calculations. The staff would not further consider the issue of CAP credit, *per se*, as a generic safety matter. The staff will update the regulatory guidance to remove the specific guidance disfavoring the use of CAP in determining NPSH margin.

(2) The staff will resume work on EPU applications, and in parallel with these reviews, the staff will conduct a backfit/regulatory analysis of the use of CAP. Depending upon the results of the backfit/regulatory analysis, the staff may backfit plants currently approved to use CAP credit and plants with current EPU applications where the applications are approved before the completion of the backfit analysis. The staff will update the regulatory guidance to reflect the results of the backfit/regulatory analysis.

Option 1 maintains the "status quo" using the current staff review procedure (including SRP Section 19.2 and no practicability assessment), except that it would use the improved guidance that resulted from ACRS recommendations to include margin and uncertainty determinations in CAP calculations. If the Commission selects Option 1, then the staff's review of license amendment requests involving CAP would not consider the remaining ACRS concerns regarding CAP.

Option 2 includes a staff action to develop a backfit/regulatory analysis of alternatives to the use of CAP credit. The backfit/regulatory analysis would determine whether eliminating some or all CAP credit represents a significant safety improvement, the costs of which could be justified.

The cost justification would be conducted consistent with NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," issued September 2004, which states that a regulatory analysis must include the following:

- the NRC's objectives for the proposed regulatory action,
- identification of and preliminary analysis of alternative approaches,
- an estimate and evaluation of the values and impacts for the selected alternatives,
- conclusion reached concerning values and impacts and a safety goal evaluation, where appropriate, and
- a decision rationale for selecting the proposed regulatory action.

The backfit/regulatory analysis would examine the value and impact of different options to resolve the CAP issue. These options would include use of CAP in determining NPSH margin and examining various plant modifications that would eliminate the need for CAP. Based on the results of the backfit/regulatory analysis, the staff would determine whether any options are justifiable from a cost/benefit standpoint using standard regulatory analysis principles, and (to the extent that an option would be imposed on plants currently approved for CAP) whether the option meets the backfit rule's cost-justified substantial increase criteria. The staff would follow the existing regulatory processes to implement the new guidance and requirements.

The staff also considered two other options that were rejected for the reasons given below:

(1) Any applications relying on CAP would be placed on hold pending the results of a backfit/regulatory analysis of the use of CAP credit. The staff rejected this option because 10 CFR 50.109(d) explicitly states that no licensing action will be withheld during the pendency of backfit/regulatory analyses.

(2) The NRC would modify current regulatory guidance by indicating that CAP credit will not be allowed for future extended power uprate (EPU) reviews. This option raises the question of why CAP can continue to be used by those plants that have increased their

power level and credited CAP. The staff rejected this option because the justification for not allowing future use of CAP credit (or an increase in the amount of CAP credit) in future EPUs would have to be justified using a regulatory analysis. Therefore, this option is subsumed by Option 2, above.

Using Option 2, the staff would continue, pending completion of the backfit/regulatory analysis, to review operating reactors and new reactors using the respective current guidance. This includes considering risk, as appropriate, consistent with SRM SECY 99-246 and SRP Section 19.2, Appendix D, for operating reactors, the application of the recently-developed deterministic staff guidance on uncertainty and pump cavitation phenomena related to NPSH margin.

RECOMMENDATION

The staff recommends Option 1 (no additional resources for fiscal years 2011, 2012, and 2013) under which the staff resumes work on EPU applications. The staff's evaluation of current EPU applications, as well as future applications for new or increased credit for CAP would be consistent with staff practice in implementing the current risk guidance on use of CAP (SRP Section 19.2) and the recently-developed deterministic guidance based on ACRS recommendations to include uncertainty and margins in CAP calculations. The staff would not consider the issue of CAP credit as a generic safety matter. This recommendation is consistent with past staff practice and is based on the following considerations as documented in Enclosure 1 to this paper.

- No regulation restricts the use of CAP. Existing regulations, staff guidance and plant technical specifications are intended to ensure that the containment is a low-leakage, robust structure, the integrity of which is demonstrated periodically (see Section 5.1 of Enclosure 1).
- ECCS and containment heat removal pumps are typically robust and have been shown to tolerate some levels of cavitation without sustaining damage (see Section 5.3 of Enclosure 1).
- The risk from allowing CAP credit for a BWR/3 with a Mark I containment with a leak detection interval of once per month is very small, based on a generic risk assessment (see Section 5.5 of Enclosure 1).
- Adequate protection of public health and safety is provided even when CAP credit is allowed.

RESOURCES:

The following staff full-time equivalent support resources are required to complete either Option 1 (the recommended option) or Option 2. The resources for Option 1 are included in the Fiscal Year (FY) 2011 Congressional Budget Justification and the FY 2012 Performance Budget request for the office of Office of Nuclear Reactor Regulation. There are no additional resources needed in the recommended option for fiscal years FY 2011, FY 2012, and FY 2013. If the Commission chooses Option 2, an additional 2.2 FTE would be required for FY 2012.

These additional funds for FY 2012 would be addressed during the FY 2013 Planning, Budgeting, and Performance Management Process (PBPM).

	Business Line	Product Line	Product	Planned Activity	Description	FY 2011		FY 2012	
						CS&T	FTE	CS&T	FTE
Option 1 (recommended)	Operating Reactors	Licensing	Licensing Actions	Other Licensing Tasks	Update Guidance Documents	0	0.6	0	1.1
Option 2	Operating Reactors	Licensing	Licensing Actions	Other Licensing Tasks	Update Guidance Documents and Perform Regulatory Analysis	0	1.0	0	3.3

<u>COORDINATION</u>:

The Office of the General Counsel has reviewed this paper and has no legal objection. The Office of the Chief Financial Officer has reviewed this paper for resource implications and concurred.

/RA by Martin J. Virgilio for/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. Use of Containment Accident Pressure
2. Areas of Agreement and Disagreement

ENCLOSURE 1

THE USE OF CONTAINMENT ACCIDENT PRESSURE

IN REACTOR SAFETY ANALYSIS

ADAMS ML102110167

THE USE OF CONTAINMENT ACCIDENT PRESSURE

IN REACTOR SAFETY ANALYSIS

CONTENTS

EXECUTIVE SUMMARY

The purpose of this paper is to discuss the technical and regulatory aspects of the use of containment accident pressure in reactor safety analysis. The discussion will encompass both presently operating reactors and new reactor designs currently undergoing staff review.

Use of containment accident pressure in determining available net positive suction head (NPSHa) has been criticized by the Advisory Committee on Reactor Safeguards (ACRS) and by participants in the U.S. Nuclear Regulatory Commission (NRC) hearing process and by members of the public. One reason for this criticism is that published regulatory guidance allowing use of containment accident pressure in determining NPSHa is not entirely consistent. The practice may also result in degradation of the regulatory philosophy of defense-in-depth (independence of fission product barriers). For these reasons, the staff has reexamined this issue. This paper presents a technical description of the use of containment accident pressure, provides a regulatory history of this issue and describes new draft staff guidance developed by the staff to quantify uncertainties and margins and address the relevant phenomena.

The use of containment accident pressure in determining the NPSHa of emergency core cooling system and containment heat removal pumps is not the subject of a regulation. Use of containment accident pressure in safety analyses (with the implicit assumption of containment integrity) is not unique to determining NPSHa. Several other important areas of safety analysis, including loss-of-coolant accident analyses and offsite radiological dose analyses assume containment integrity.

The staff has accepted predictions of pump operation in cavitation for a limited time based on testing under cavitation conditions for the time period that the pump is predicted to cavitate and when post-test inspections verify that no damage resulted from such operation. This is consistent with the guidance of Regulatory Guide (RG) 1.82, Revision 3.[1] However, the staff reexamined this situation and developed new draft guidance with more stringent conditions under which this practice is acceptable.

Published at different times, the regulatory guidance on the issue of using containment accident pressure in determining NPSHa is inconsistent. The position of Regulatory Guide 1.82 Revision 3 is that the use of containment accident pressure in determining NPSHa for operating reactors is acceptable “where the design cannot be practicably altered.” Earlier published guidance (Regulatory Guide 1.1[2]) does not permit this use. Revision 5 of Standard Review Plan (SRP) Section 6.2.2[3] reflects the position of RG 1.82 Revision 3, and includes acceptance criteria for cases where containment accident pressure is used. Revision 5 of Section 6.2.2 of the Standard Review Plan also discusses the use of risk to justify use of containment accident pressure for containment spray pumps but does not mention the emergency core cooling system pumps. No other staff guidance mentions risk as a consideration in evaluating the use of containment accident pressure. Following the NRC staff’s completion of a regulatory analysis, the staff plans to make these regulatory guidance documents consistent.

An important portion of the NRC staff review of fires under Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,”[4] to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,“ deals with the spurious operation of equipment. This spurious operation is

caused by hot shorts, open circuits or shorts to ground in associated nonsafety circuits in fire areas. As part of the staff review of a proposed use of containment accident pressure in determining NPSHa, the staff ensures that spurious operation of equipment does not adversely affect containment isolation.

In response to an ACRS recommendation, the staff has developed draft guidance that includes the determination of the margin and uncertainty associated with the use of containment accident pressure in determining NPSHa. This draft guidance has been discussed with ACRS and the Boiling Water Reactor Owners' Group.

At present time, 60 pressurized water reactors and 19 boiling water reactors credit containment accident pressure to demonstrate adequate net positive suction head margin.

The staff plans to apply the same criteria to new reactor designs as those that are currently applied to operating reactors.

THE USE OF CONTAINMENT ACCIDENT PRESSURE IN REACTOR SAFETY ANALYSIS

1.0 CONTAINMENT ACCIDENT PRESSURE

The purpose of this enclosure is to discuss the technical and regulatory aspects of the use of containment accident pressure in reactor safety analysis. The discussion will encompass both presently operating reactors and new reactor designs currently undergoing review by the staff of the U.S. Nuclear Regulatory Commission (NRC).

Before proceeding, it is necessary to understand what is meant by containment accident pressure and the use of containment accident pressure or credit for containment accident pressure.

The initial pressure in light water reactor containments is maintained close to atmospheric pressure except for the subatmospheric containment, which is maintained at a partial vacuum.

The licensing basis of every operating reactor contains a set of postulated accidents. The licensee for each reactor must demonstrate that, for each of these accidents, the respective safety criteria are satisfied. Some postulated accidents are initiated by the rupture of a high-energy pipe within the containment, a reactor coolant system pipe or a steamline. The discharge of high-pressure, high-temperature water or steam from these high-pressure systems after rupture results in the rapid heating of the containment atmosphere. This rapid heating increases the pressure (and temperature) of the containment atmosphere. The pressure in containment during a postulated accident is the containment accident pressure. The total containment accident pressure is made up of two partial pressures: the partial pressure of the water vapor and the partial pressure of the heated air or nitrogen present in the containment atmosphere before the postulated accident.

Use of or credit for containment accident pressure in reactor safety analyses means that this pressure is included in demonstrating that one or more safety criteria are satisfied. Since this pressure is, for high-energy pipe ruptures, greater than the containment pressure before the accident, crediting this pressure implies that containment integrity is maintained.

Containment integrity, in this case, means that the containment is capable of maintaining whatever containment accident pressure is being included in the safety analyses to demonstrate that a safety criterion is satisfied. Analyses typically assume that the containment is leaking at its maximum leakage rate, L_a, as defined in the regulations.

As discussed later in this paper, for many pressurized water reactors (PWRs), the containment pressure during an accident is assumed to be the vapor pressure at the temperature of the sump water. Some safety analyses conservatively ignore the partial pressure of the air in containment. However, if this vapor pressure is greater than the pressure in containment before the accident, it is considered containment accident pressure.

An earlier term used for this issue was “containment overpressure.” The staff discontinued use of this term for two reasons: (1) the industry uses several definitions of containment overpressure, and (2) it has been confused with overpressurizing the containment. The containment design pressure is never exceeded while crediting containment accident pressure.

Currently, the staff allows credit for containment accident pressure in analyzing three situations:

1. to demonstrate adequate available net positive suction head (NPSH) for the emergency core cooling system (ECCS) and containment heat removal system pumps
2. to demonstrate that flashing (phase change from water to steam) will not occur across the sump screens in a PWR
3. to demonstrate that flashing (phase change from water to steam) will not occur in the bulk sump water from too low an internal pressure. This could uncover a portion of the sump screen and affect pump suction conditions

Items 2 and 3 affect the pump suction conditions and therefore, the available NPSH. Thus, they could be considered as included in item 1 but are listed separately for clarity.

Since sufficient containment accident pressure for each of these situations implies the integrity (leak tightness) of the containment, the regulatory philosophy of defense-in-depth (independence of fission product barriers) could potentially be challenged.

2.0 CAVITATION AND NET POSITIVE SUCTION HEAD

A major consideration in the design and operation of centrifugal pumps and the systems in which they operate is cavitation. Cavitation is the generation of vapor bubbles in a liquid as a result of a reduction in local static pressure below the vapor pressure of the liquid and the subsequent collapse of these vapor bubbles as the liquid flows through the pump from a region of relatively low pressure to a region of higher pressure.

Centrifugal pumps are subject to three types of cavitation:

(1) vaporization cavitation,
(2) internal recirculation cavitation and
(3) vane passing syndrome cavitation.

Table 1 explains these types further, but this paper is concerned only with vaporization cavitation. The two other types of cavitation either are associated with flows less than those important to the functioning of the ECCS and containment spray pumps during an accident (internal recirculation) or are a sensitive function of the pump design (vane passing cavitation); for these reasons, they will not be discussed further.

The pump inlet is called the pump suction, and the pump outlet is called the pump discharge. Although the inlet is termed the suction side, a centrifugal pump does not “suck” liquid into the pump. Rather, the liquid must be forced into the eye of the pump (the point of minimum radius of the impeller with reference to the pump centerline) where it enters the blade passages of the impeller. The liquid is forced into the pump by the pressure difference between the pressure at the pump suction flange and the pressure at the eye of the impeller. To avoid cavitation, it is necessary to provide a sufficiently high inlet pressure on the liquid so that the lowest pressure in the pump remains above the liquid vapor pressure.

The NPSH is the difference between the stagnation pressure at the pump suction and the liquid vapor pressure. It is, therefore, a measure of the energy forcing the liquid into the pump. There are two related quantities. The NPSHa is a function of the piping system design, the pump flow rate and the temperature of the pumped water. The required NPSH (NPSHr) is determined by measurement for a given pump and is the NPSH that produces a given (acceptable) amount of cavitation (measured as reduction in pump discharge head). NPSHr is a function of the pump design and pump flow rate. The NPSHr increases as the flow rate increases. Because of higher flow losses, the NPSHa decreases as the flow increases.

When the suction pressure or the NPSHa is decreased from the value corresponding to cavitation inception, the region of cavitation enlarges and, if the decrease in NPSHa is sufficient, noise, cavitation erosion of pump parts (mainly the impeller), and pump performance degradation will occur.

Table 1 Types of Cavitation

CAVITATION TYPE	CAUSE	EFFECT ON PUMP
Vaporization	The pressure at the low- pressure point in the pump is below liquid vapor pressure. The vapor bubbles formed because of this low pressure collapse (implode) after entering a higher pressure region of pump.	(1) erosion (pitting) of impeller (2) reduction in flow and head (3) damage to seals and bearings (4) excessive noise and vibration
Internal Recirculation	Reverse flow occurs at the pump suction.	(1) erosion (pitting) of impeller at a different location than vaporization cavitation (2) reduction in flow and head (3) damage to seals and bearings
Vane Passing Syndrome	The clearance between the impeller blades and cutwater is too narrow; this allows high-velocity backflow that reduces pressure and causes local vaporization.	(1) damage to impeller tip (2) loss of efficiency and pressure fluctuations

For satisfactory long-term operation of a centrifugal pump, the NPSHa should exceed the NPSHr. The difference between NPSHa and NPSHr is the NPSH margin. As the liquid flows through the impeller, a point is reached where the pressure has increased sufficiently so that any vapor bubbles present in the flow as a result of vaporization will collapse (implode) causing small but intense shock waves in the liquid. The cumulative effect of the shock waves from thousands of collapsing bubbles may include erosion of the pump impeller as well as other detrimental effects. (The pressure resulting from the implosive collapse of one bubble is estimated to be on the order of 10^4 atmospheres). The flow imbalance caused by the vapor formation and collapse can damage the pump shaft, seals, and bearings. The presence of vapor in the flow also degrades the flow rate and discharge head. The formation of vapor results in a reduction in the capacity of the pump, since an increase in the volume of the vapor causes a decrease in the volume of liquid. The bubbles, unlike the liquid, are highly compressible, and therefore, the head developed by the pump is reduced. In addition, energy is expended increasing the velocity of the liquid filling the cavities as the vapor collapses. This further reduces the discharge head of the pump.[5]

The NPSHa can be determined from the following equation for the simple flow path shown in Figure 1 of a pump taking suction from a closed tank.

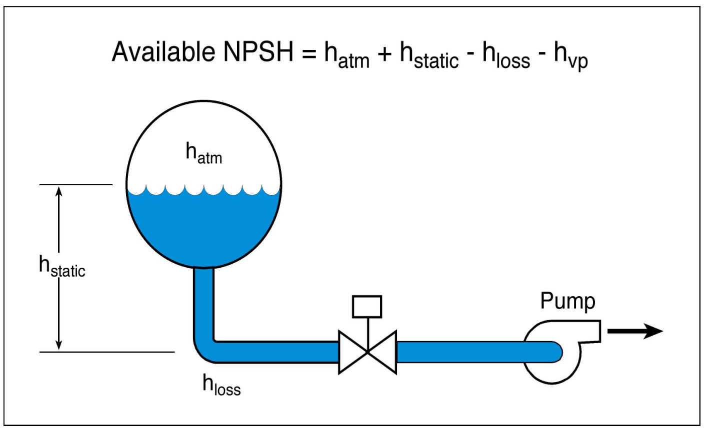


Figure 1 Determination of NPSHa

$$NPSHa = h_{atm} + h_{static} - h_{loss} - h_{vp}$$

where

h_{atm} = the head on the liquid surface resulting from the pressure in the atmosphere above

h_{static} = the head resulting from the difference in elevation between the liquid surface and the centerline of the pump suction

h_{loss} = the head loss resulting from fluid friction and fittings in the flowpath to the pump suction flange

h_{vp} = the head equivalent to the vapor pressure of the water at the water temperature

h_{atm} would be the head resulting from atmospheric pressure if the tank is open to the atmosphere. Pressurizing the tank increases h_{atm} and therefore, increases the pump NPSHa. If one imagines the tank to be the suppression pool of a boiling water reactor (BWR) or the water on the containment floor of a PWR, the pressure of the atmosphere above the liquid surface is the pressure of the containment atmosphere. During some postulated accidents, the pressure in the containment will increase because of the discharge of steam and flashing hot water into the containment. Because of conditions adverse to NPSH margin, such as increased temperature of the water in the sump or the suppression pool or increased head loss at the pump suction screens due to debris blocking the screens, sufficient NPSH margin may not be available in some cases if containment accident pressure is not available.

The NPSHa for many PWRs is determined with the assumption that the pressure at the liquid surface (h_{atm}) is equal to the vapor pressure (h_{vp}) of the liquid; that is, the following equation applies:

$$NPSHa = h_{static} - h_{loss}$$

This approach ensures that partial pressure resulting from the air (or nitrogen) in containment above the liquid surface is not included in determining NPSHa and the NPSHa value is therefore, conservative. However, when the temperature of the sump water is greater than 212 degrees F, the vapor pressure will be greater than the pressure in containment before the postulated accident and containment integrity is assumed.

NPSHr is defined as that value of NPSH that results in a 3-percent drop in pump discharge head.[6] This value of NPSHr is denoted $NPSHr_{3\%}$. The 3-percent value is used for two reasons. First, the value is relatively easy to determine by testing. Second, most standard low suction energy pumps can operate with little or no margin above this NPSHr point without seriously affecting the pump's long-term operation. However, the full published pump head is not achieved when the NPSHa equals this value of NPSHr. The head is 3-percent less than the fully developed head. It can take from 1.05 to 2.5 times this NPSHr value to achieve the 100-percent discharge head.

Normal practice in pump operation requires that the NPSHa be greater than the NPSHR by some margin. The value of this margin differs from one authority to another. Some experts state that using a value of NPSHa equal to the NPSHr is acceptable. The nuclear industry practice has been to determine the amount of containment accident pressure needed so that the NPSHa equals the NPSHr with no margin specified. This practice is acceptable for the because the calculated containment pressure for the loss-of-coolant accident (LOCA), as discussed below, is conservatively calculated and therefore, contains margin. Also, the uncertainty in NPSHr is included in the calculation as explained in Section 6.0, "Staff Guidance."

Suction conditions in a centrifugal pump can be characterized by the suction specific speed which is defined as the following:

$$N_{SS} = \frac{n\,Q^{1/2}}{(NPSHr_{3\%})^{3/4}}$$

where

n = the pump speed in revolutions per minute (rpm)

Q = the pump capacity (volumetric flow rate) in gallons per minute (gpm)

NPSHr = the required NPSH in feet (ft)

Both Q and NPSHr are taken at the best (maximum) efficiency point on the pump curves. Values of N_{ss} for BWR ECCS and core spray pumps can be in the range of 12,000 or more.

The suction energy concept is also used to assess the suction capability of a pump. Suction energy is defined as the following:

$$\text{suction energy} = D_{eye}\ n\ N_{ss}\ sg$$

where

D_{eye} = pump eye diameter (inches)

n = pump speed (rpm)

N_{ss} = suction specific speed defined above

sg = specific gravity

Based on experience with hundreds of centrifugal pumps, specific gravity values have been derived for the start of “low suction energy,” “high suction energy,” and “very high suction energy,” for various centrifugal pump types. Suction energy was proposed as an alternative to suction specific speed in specifying acceptable suction conditions since it was found that pump suction specific speed was not always a dependable parameter to differentiate acceptable from unacceptable regions of operation with respect to cavitation.[7]

Another pump parameter mentioned in this paper is the total dynamic head. It can be thought of as the total energy delivered to the pumped liquid or as the equivalent height that a fluid is to be pumped.

3.0 LIGHT-WATER REACTOR SAFETY SYSTEM DESIGN

The ECCS and the containment spray system portion of the containment heat removal system for both BWRs and PWRs use centrifugal pumps to supply water to the reactor core and containment spray spargers.

3.1 Pressurized Water Reactors

For the PWR, the design basis LOCA is the only postulated event that may require use of accident pressure in determining the NPSHa of the ECCS or containment spray pumps. Following a reactor coolant system pipe break (LOCA) in a PWR, a two phase mixture of steam and water is discharged into the containment atmosphere. This rapidly increases the containment pressure. An engineered safety feature actuation signal, based on loss of water from the reactor vessel or the increase in containment pressure, actuates the ECCS and the containment spray system.

In a PWR, the ECCS pumps initially take suction from a large tank of borated water (the refueling water storage tank (RWST)) and inject this borated water into the reactor core to keep the nuclear fuel cool. The water in the RWST is typically at a temperature close to ambient and remains at this initial temperature. When the water in the RWST reaches a low level, the suction of each pump transfers either automatically or by operator action to an emergency sump located within the containment. This water is injected into the reactor vessel, spills from the pipe break to the sump and is recirculated back to the reactor vessel. The water on the containment floor and in the sump consists of reactor coolant, water from the RWST, and ECCS accumulators. The residual heat of the reactor coolant system and the decay heat generated by the nuclear fuel heats the water in the sump to a high temperature (most operating PWRs have maximum sump water temperatures well in excess of 212 degrees F). Screens upstream of the pumps filter out most of the debris in the water (e.g., thermal insulation on piping and paint on the containment walls) generated by the pipe break. Typically, the sump temperature is at its maximum value at the beginning of recirculation. The containment spray system reduces the containment pressure using water from the RWST and the containment sump. The reduction in containment pressure helps to maintain the containment pressure within design limits and limits the calculated release of radioactivity to the environment.

In PWR accident analysis, the recirculation phase of the LOCA is usually most limiting for ECCS and containment spray pump NPSH since the pumped water is at high temperature and the water may contain debris generated by the accident, which can accumulate on sump screens or strainers and cause an increase in the head loss.

The new reactor PWR designs (active only, not passive) are very similar in design and operation to the existing PWRs. One main difference is that the ECCS takes suction from an in-containment RWST located at the bottom of the containment. This in-containment water storage tank serves as the containment emergency sump; switchover from an outside-containment water storage tank is no longer necessary. The in-containment tank has the maximum water level at the start of the accident. The tank’s level lowers during the accident because of water holdup on the floors, surfaces and compartments that do not return water to the storage tank/emergency sump.

3.2 Boiling Water Reactors (BWRs)

In BWR accident analysis, several postulated accidents, in addition to the LOCA, can result in heating of the suppression pool, which could be limiting with respect to ECCS pump NPSH. The LOCA produces high temperatures in the suppression pool water and generates debris that can accumulate on the ECCS pump suction strainers and cause an increase in head loss. Both of these effects reduce the NPSHa. Several other postulated BWR accidents are also considered. These are the plant response to a fire postulated according to the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.48, “Fire Protection,” and Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities” (hereafter referred to as the “Appendix R Fire”); the plant response to a station blackout postulated according to the requirements of 10 CFR 50.63, “Loss of All Alternating Current Power”; and the plant response to anticipated transients without scram (ATWS) postulated according to the requirements of 10 CFR 50.62, “Requirements for the Reduction of Risk from ATWS Events for Light-Water-Cooled Nuclear Power Plants.” While these events result in high suppression pool temperatures, they are not generally accompanied by the generation of debris that could clog ECCS pump suction strainers. (Some older BWRs have separate safety valves that discharge into the drywell and could generate debris).

The currently operating BWRs have one of three containment designs. The use of containment accident pressure for NPSHa of the low head ECCS pumps is limited to the Mark I design, which is the earliest design still in operation.

The Mark I containment consists of a drywell and a pressure suppression chamber, also called the torus or the wetwell. Technical specifications require that, during operation, the air in the drywell and the wetwell atmospheres is replaced with nitrogen (called inerting) to eliminate the potential for hydrogen combustion in the containment in the event of its release from the vessel during an accident or its formation by radiolysis. The drywell is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion that contains the reactor coolant system. The pressure suppression chamber is a large toroidal-shaped vessel partially filled with a large amount of water located within the primary containment. It serves several purposes. The suppression pool will condense the steam released by a reactor coolant system pipe break during the postulated LOCA or the steam discharged from the reactor vessel safety/relief valves. It also serves as the water source for the low-head ECCS pumps. The drywell is connected to the pressure suppression chamber or torus by a vent system that conducts flow without excessive resistance and distributes the flow uniformly in the suppression pool following a postulated LOCA.

Following a break of a recirculation line in the reactor coolant system (the postulated LOCA) the drywell pressure rapidly increases and the water initially in the submerged portion of the vent system accelerates into the suppression pool. Until the water is completely cleared from the vent system, the nitrogen pressure in the wetwell does not increase significantly. Flow to the wetwell is initially nitrogen but subsequently becomes a mixture of nitrogen, steam and water. At first, the drywell pressure increases more rapidly than the wetwell pressure. Flow through the vent system begins to decrease and the drywell and wetwell pressures converge and differ only by the static pressure resulting from the vent system's submergence in the suppression pool. The suppression pool water is heated during the accident first, as a result of the condensation of steam released from the pipe break and subsequently by the residual heat in the reactor coolant system and the decay heat generated in the core. The action of the ECCS pumps can also generate significant heat. Strainers upstream of the BWR low head ECCS pumps filter out debris generated by the pipe break that flows into the suppression pool. The suppression pool water circulates from the suppression pool to the reactor vessel and subsequently flows out the break and returns to the suppression pool.

BWR Mark I containment spray systems are a portion of the residual heat removal (RHR) system and are manually actuated. Both drywell and wetwell sprays cool the respective atmospheres following a LOCA to reduce the containment pressure.

The following three subsections discuss three "special events" that also may need credit for containment accident pressure in BWRs because they also result in heating the suppression pool water. These three special events are the Appendix R fire, station blackout and the anticipated transient without scram (ATWS). Since these three events do not result in recirculation in a PWR, PWR safety analyses do not have to consider containment accident pressure for these events.

3.2.1 Appendix R Fire

A postulated fire consistent with Appendix R to 10 CFR Part 50 (Appendix R) is intended to limit fire damage to systems required to achieve and maintain safe shutdown. The regulations place stringent limits on the damage allowed to those systems required to safely shut down the plant, with the intent of ensuring that protection is provided so that a fire within one system will not damage the redundant system.

With the exception of the loss of offsite power considerations for some fire areas, other accidents or events are not assumed to occur unless caused by the fire. For an Appendix R fire, this provides for the containment response to be consistent with a typical transient. While the monitoring of containment pressure is expected, the potential to overpressurize containment is not increased when compared to the design-basis accident (DBA). However, the potential exists for other concerns, based on the location and magnitude of the fire, to cause the misoperation or maloperation of equipment. Various regulatory requirements are in place to ensure that at least one train of equipment remains free from fire damage. The potential for misoperation or maloperation of equipment is addressed in both Appendix R and NFPA 805.[8]

The fire protection program includes an analysis to demonstrate that the systems, structures and components (SSCs) important to safety can accomplish their respective postfire safe-shutdown functions. The safe-shutdown analysis is used to demonstrate that redundant safe-shutdown systems and components, including electrical circuits for which fire-induced failure could directly or indirectly prevent safe shutdown, are adequately protected such that one success path remains free of fire damage in a postulated fire. This analysis includes various conservative assumptions, including the expectation of an exposure fire (defined in 10 CFR Part 50, Appendix R) and the failure of all equipment in the associated fire area.

Once the success path has been determined, procedures are developed from this analysis. For fire areas that contain redundant equipment, but have fire barriers or physical separation demonstrated to be sufficient (see Appendix R, Section III.G.2, concerning redundant shutdown), the analysis ensures that a dedicated set of equipment is available for safe shutdown. In this case, licensees may use fire-related operating procedures, plant emergency operating procedures, or other abnormal operating procedures, which can be event-based. Given the protection of one train of equipment in the event of a fire, these symptom- or event-based response strategies are fully capable of maintaining the plant in a safe condition. The strategies in use include the monitoring and control of various plant conditions, including containment pressure and temperature, and wetwell pressure and temperature (for BWRs), as the plant is shut down using available equipment.

From the safe shutdown analysis, other procedures are also developed that describe the means to implement a shutdown from outside the control room (see Appendix R, Section III.G.3, concerning alternative or dedicated shutdown capability). Consistent with the requirements of Section III.L of Appendix R, these procedures must address shutdown when offsite power is available and when offsite power is not available for 72 hours. These procedures are required to address necessary actions to compensate for spurious actuations and high-impedance faults if such actions are necessary to effect safe shutdown. Consistent with Section III.L(2) of Appendix R, several performance goals have been established, including ensuring that the reactor heat removal function is capable of achieving and maintaining decay heat removal.

The conservative regulatory assumptions put in place to ensure that mitigation equipment remains available to shut down the plant, can result in the need for use of containment accident pressure to ensure adequate NPSH margin. As the regulations require that design regulatory analyses be conservative, the adequacy of NPSH margin must be assessed for each fire area that uses the alternative dedicated shutdown capability.

3.2.2 Station Blackout

A station blackout is the complete loss of alternating current electrical power to the essential and nonessential switchgear buses in a nuclear power plant. As stated in 10 CFR 50.63, a light water reactor "must be able to withstand for a specified duration and recover" from a station blackout.

For a BWR with a Mark I containment, the reactor will scram, the turbine generator will trip, and the reactor vessel will be isolated by closure of the main steam isolation valves (MSIVs) following a station blackout. The reactor scram and the pressure increase resulting from closure of the MSIVs will result in a drop in the reactor vessel water level. The operator will respond by initiating both the high-pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system to restore the reactor vessel water level. Decay heat generation will cause the reactor vessel pressure to increase to the setpoints of as many safety/relief valves as are required to terminate the pressure increase. Pressure in the reactor vessel will be maintained by periodic blowdown through the safety/relief valves to the pressure suppression pool. The steam will be condensed, and the temperature of the suppression pool water will monotonically increase since there will be no means of suppression pool cooling under station blackout conditions. The HPCI and RCIC systems take suction from the condensate storage tank and are therefore, independent of the suppression pool. However, they have the capability to also take suction from the suppression pool. The HPCI and RCIC pumps are turbine driven by steam taken from the main steam piping upstream of the MSIVs. Operation of these systems will maintain the reactor in a stable cooled state. The operation time of these systems depends on the capacity of the unit batteries.

After restoration of a source of alternating current power, suppression pool cooling by the RHR pumps can be initiated, and the suppression pool temperature will monotonically decrease thereafter. The analysis of this event must show that adequate NPSHa exists for the RHR pumps to continue to cool the suppression pool once alternating current power is restored.

3.2.3 Anticipated Transient without Scram (ATWS)

An ATWS is defined in 10 CFR 50.62 as an anticipated operational occurrence (defined in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," as a condition of normal operation that occurs one or more times during the lifetime of the plant), followed by the failure of the reactor trip portion of the reactor protection system. For the BWR, the most limiting ATWS event is typically the MSIV closure without scram. This event typically results in a higher peak pressure and a larger amount of steam discharged to the suppression pool than in any other ATWS event. Closure of the MSIVs isolates the turbine and condenser from the pressure vessel. The main condenser is no longer available to condense steam, and the feedwater pumps and feedwater heaters will not function if steam-driven. The closure of the MSIVs causes a rapid pressurization of the reactor pressure vessel, which collapses steam voids and increases reactor power. Loss of the feedwater heaters also contributes to a higher power level. Vessel pressure is relieved entirely by the safety/relief valves. Steam is directed through spargers called "T-quenchers." The T-quenchers are designed to promote complete condensation in the pool. This condensation heats the suppression pool. Part of the steam is diverted from the main steamline to drive the HPCI and RCIC turbines, which provide high-pressure makeup water to the vessel. The control rod drive pumps can also contribute to vessel water makeup as they take water from the condensate storage tank. The high reactor vessel pressure initiates a recirculation pump trip. The lower flow also adds negative reactivity, reducing reactor power. The operator further introduces negative reactivity by manually initiating injection of boron by means of the standby liquid control system.

After reactor coolant pump coastdown following recirculation pump trip, the reactor power stabilizes at a power level dependent on the scenario. The steaming rate to the suppression pool is slightly less than this value. The difference is the energy needed to heat the subcooled ECCS fluid to saturation.

The operator locks open the safety/relief valves to control vessel pressure in order to minimize safety/relief valve cycling. Since the condenser is not available for this event, the turbine bypass valves cannot be used for this purpose.

When the suppression pool temperature exceeds 95 degrees F, the emergency operating procedures require the operator to monitor and control drywell temperature, suppression pool temperature, and primary containment pressure. When the suppression pool temperature exceeds 95 degrees F, the operator is instructed to initiate suppression pool cooling.

At some point, the reactor vessel level passes the triple low water level setpoint because of the termination of feedwater. The RHR pumps are then automatically aligned for the low-pressure coolant injection (LPCI) mode of operation and interlocked on a 5-minute timer. This prohibits realignment of the RHR system for suppression pool cooling, even though LPCI is not needed for this event. The timer is a LOCA consideration to ensure adequate core cooling. After the 5-minute timer restriction has ended, the operator may align the RHR system for suppression pool cooling, assuming that the level does not again fall below the triple low level setpoint, and reactuate the timer.

The operator must maintain the suppression pool temperature below the heat capacity temperature limit specified in the emergency operating procedures. This is a curve of suppression pool temperature as a function of reactor pressure vessel pressure. If the suppression pool temperature cannot be maintained below this curve for a given reactor pressure, the reactor vessel must be depressurized to prevent containment failure if emergency depressurization would be required later in the transient when the suppression pool might be at a higher temperature and less able to condense steam. The depressurization produces steam flashing, and the reactor power is significantly reduced. The reactor may even be shut down. However, when the depressurization is complete, criticality may again be achieved.

The injected boron reduces the reactor power, and suppression pool cooling controls the suppression water temperature. The suppression pool water temperature affects the NPSHa of the RHR pumps.

4.0 REGULATORY POSITION ON USE OF CONTAINMENT ACCIDENT PRESSURE IN DETERMINING AVAILABLE NPSH

4.1 Defense-in-depth

Various NRC documents describe defense-in-depth as a concept or a philosophy. The NRC Strategic Plan defines defense-in-depth as "an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or lessen the effects of damage if a malfunction or accident occurs at a nuclear facility." The glossary on the NRC's public Web site offers the following definition of defense-in-depth:

> "An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures."

With the exception of a few specific applications, defense-in-depth is not included in 10 CFR Part 50. The exceptions are 10 CFR 50.48 and Appendix R to 10 CFR Part 50, which characterize defense-in-depth in terms of fire protection: prevention, detection, suppression, mitigation and safe-shutdown capability; and 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors." 10 CFR 50.69 mentions defense-in-depth but does not characterize it for that application. The regulation in 10 CFR Part 100, "Reactor Site Criteria," mentions defense-in-depth in regards to the siting of plants. Additional regulations that use the phrase "defense-in-depth" include 10 CFR 70.64,[9] 10 CFR 73.54, 10 CFR 73.55, and Appendix C to 10 CFR Part 73.

Fission product barriers (or "safety barriers") are mentioned in 10 CFR 50.34; 10 CFR 50.36; 10 CFR 50.59; 10 CFR 50.72; Appendix A to 10 CFR Part 50; Appendix J to 10 CFR Part 50; and 10 CFR Part 100. In Appendix A to 10 CFR Part 50, Criteria 10 through 19 are grouped under the heading "Protection by Multiple Fission Product Barriers." The concept of independence is included in Criterion 17 ("Electric Power Systems"), Criterion 21 ("Protection System Reliability and Testability"), Criterion 22 ("Protection System Independence"), Criterion 24 ("Separation of Protection and Control Systems"), and Criterion 26 ("Reactivity Control System Redundancy and Capability"). However, the regulations do not discuss the degree or level of independence among fission product barriers.

In fact, the fission product barriers are not independent for all possible accidents. For DBAs, the containment is assumed to be intact; all DBAs are analyzed assuming containment integrity. Containment integrity must be shown to be maintained for certain postulated events that are not considered DBAs. These include postulated ATWSs and station blackout events. Thus, the assumption of loss of containment integrity (containment failure) is a beyond-design-basis event.

Containment integrity for DBAs is maintained if the containment leakage rate is less than or equal to L_a. Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50 defines L_a as the maximum allowed mass leakage rate (per 24 hours) at the calculated peak containment internal pressure related to the design-basis LOCA. Each plant's technical specifications include the value of L_a.

Several regulations assume containment integrity. For example, 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," states that the calculated containment pressure should be minimized in LOCA calculations. The assumption is that containment pressure is present but minimized. The offsite dose limits of 10 CFR Part 100 or 10 CFR 50.67, "Accident Source Term," could not be met without containment integrity.

The interfacing systems LOCA (ISLOCA), which is not a design-basis event, and the steam generator tube rupture, which is a design-basis event, both involve bypassing the containment. The ISLOCA is acceptable based on low risk based on plant-specific analyses. The steam generator tube rupture is acceptable based on acceptable offsite dose.

The topic of using risk information in determining the degree of defense-in-depth has been brought to the Commission's attention before in a May 19, 1999, letter from Dana A. Powers, ACRS Chairman, to NRC Chairman Shirley Ann Jackson. The letter points to a paper titled "On the Role of Defense-in-depth in Risk-Informed Regulation," presented at the PSA '99 conference, held August 22–25, 1999. Together, the letter and the attached paper illustrate two different but not conflicting philosophical approaches to models of the scope and nature of defense-in-depth. The structuralist model asserts that defense-in-depth is embodied in the structure of the regulations and in the design of the facilities built to comply with those regulations, regardless of any assessment of probability. The rationalist model asserts that defense-in-depth is the aggregate of provisions made to compensate for uncertainty and incompleteness in our knowledge of accident initiation and progression. This model is made practical by the development of the ability to quantify risk and estimate uncertainty using probabilistic risk assessment (PRA) techniques. The paper mentions that both models can be construed as a means of dealing with uncertainty, and neither incorporates any reliable means of determining when the degree of defense-in-depth achieved is sufficient. The fundamental difference is that the structural model accepts defense-in-depth as fundamental, while the rationalist model places defense-in-depth in a subsidiary role. The paper recommends a hybrid approach between the two models in which the structuralist model of defense-in-depth would be retained as the high-level safety philosophy and the rationalist model would be used at lower levels in the safety hierarchy. The level at which the models switch roles would be a policy issue.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,"[10] states the following:

> …an engineering evaluation should evaluate whether the impact of the proposed licensing basis change is consistent with the defense-in-depth philosophy. If a comprehensive risk analysis is done, it can be used to help determine the appropriate extent of defense-in-depth (e.g., balance among core damage prevention, containment failure, and consequence mitigation) to ensure protection of public health and safety. When a comprehensive risk analysis is

not or cannot be done, traditional defense-in-depth considerations should be used or maintained to account for uncertainties.

The statement above highlights the rationalist model, in which defense-in-depth takes a subsidiary role. NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition", Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," issued June 2007, adds the following to the above statement: "However, because PRA does not reflect all aspects of defense-in-depth, appropriate traditional defense-in-depth considerations should also be used to account for uncertainties."

This suggests that PRAs have an inherent uncertainty which they may or may not be able to depict.

In addition, RG 1.174 states that consistency with the defense-in-depth philosophy is maintained if the following conditions are met:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.

- Overreliance on programmatic activities to compensate for weaknesses in plant design is avoided.

- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

- Defenses against potential common-cause failures (CCFs) are preserved, and the potential for the introduction of new CCF mechanisms is assessed.

- Independence of barriers is not degraded.

- Defenses against human errors are preserved.

- The intent of the general design criterion (GDC) in Appendix A to 10 CFR Part 50 is maintained.

The fifth bullet above maintains a structuralist viewpoint. Credit for containment accident pressure may degrade defense-in-depth, since avoiding core damage may depend on containment integrity.

4.2 Regulatory Background and Current Practice

As part of reactor safety calculations, licensees must demonstrate that the ECCS pumps and containment heat removal pumps will perform their safety function of delivering "abundant flow," as required by 10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling," and rapidly reducing the containment pressure and temperature, as required by 10 CFR Part 50, Appendix A, GDC 38, "Containment Heat Removal." The ECCS pumps must perform their safety function during a LOCA, in order to satisfy the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

Some of the first nuclear power plants licensed in this country included in their design the use of containment accident pressure in the determination of NPSHa for the ECCS pumps. ACRS identified this issue during the review of several BWRs in the late 1960s.[11] To resolve this concern, in November 1970, the Atomic Energy Commission issued Safety Guide 1 (now RG 1.1, "Net Positive Suction Head for Emergency Core Cooling System and Containment Heat Removal System Pumps," issued November 1970).

NRC regulatory guides are issued to describe and make available to the public methods acceptable to the NRC regulatory staff for implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems and to provide guidance to the nuclear industry. Regulatory guides are not themselves regulations and compliance with them is not required. The staff may find methods and solutions different from those set out in the regulatory guides acceptable if they provide a basis for issuance of a license or a license amendment.

RG 1.1, which remains part of the licensing basis of some operating reactors, contains a single position, which states the following:

> Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and <u>no increase in containment pressure from that present prior to postulated loss-of-coolant accidents</u>. (emphasis added)

The NRC allowed use of containment accident pressure in determining NPSHa for some reactors that received their construction permits before the issuance of RG 1.1. The licensing bases of BWRs receiving their construction permits after issuance of RG 1.1 were generally consistent with its guidance. Many PWRs use the assumption that the containment pressure is equal to the sump water vapor pressure, which in most cases is greater than the pressure in containment before the postulated accident. This is inconsistent with RG 1.1 (and therefore, assumes containment integrity) but results in a conservative value of NPSHa since it ignores the partial pressure of the air in containment. Some operating reactors continue to include RG 1.1 in their licensing basis.

The NRC licensed PWRs with subatmospheric containments using containment accident pressure to determine NPSHa. Because the subatmospheric containment design requires that containment accident pressure be reduced below atmospheric pressure within one hour following a design-basis LOCA, these plants are provided with a recirculation spray system that takes suction from the sumps earlier in the LOCA than the switch to recirculation for other PWRs (i.e., while water remains in the RWST). Since the recirculation spray pumps take suction from the sump early in the event, the sump water level is lower, and use of containment accident pressure is necessary for adequate NPSH margin. Revision 4 to SRP Section 6.2.2.II.2 states that, for subatmospheric containments, "the guidelines of Regulatory Guide 1.1…will apply after the injection phase has terminated…. Prior to termination of the injection phase the NPSH analyses should include conservative predictions of the containment atmosphere pressure and sump water temperature transients."

On December 3, 1985, the NRC issued Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage,"[12] which discussed the findings and resolution of NRC Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." USI A-43 dealt with ECCS and containment spray pump suction issues. One of these issues was the blockage of ECCS sump screens in PWRs and, to a lesser extent, the potential for blockage of BWR suction strainers. NUREG-0897, Revision 1, "Containment Emergency Sump Performance, Technical Findings Related to Unresolved Safety Issue A-43," issued October 1985, documents the technical findings of USI A-43. NRC GL 85-22 discussed these findings, which include the fact that (1) blockage of sump screens by LOCA-generated debris requires a plant-specific resolution, and (2) a revised screen blockage model should be applied to emergency sump screens. However, the NRC regulatory analysis of this issue resulted in the decision not to backfit these findings.[13] GL 85-22 recommended that licensees use the technical guidance developed while studying this issue for any future modifications to thermal insulation inside containment or to primary coolant system piping.

As part of the resolution of USI A-43, the NRC revised SRP Section 6.2.2 (Revision 4)[14] to include guidance on NPSH. The revised guidance stated, "The NPSH analysis will be acceptable if (1) it is done in accordance to the guidelines of NUREG-0897 and (2) it is done in accordance with the guidelines of Regulatory Guide 1.1." Thus, after this first examination of the effects of LOCA-generated debris on the NPSHa of ECCS pumps, the guidance for calculating NPSHa remained that found in RG 1.1.

On July 28, 1992, the Barsebäck Unit 2 BWR in Sweden experienced a spurious opening of a pilot-operated relief valve at 435 pounds-force per square inch gauge. This resulted in the dislodging of mineral wool insulation, which subsequently blocked emergency pump suction strainers. The reactor was safely shut down. Subsequently, the NRC issued Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. All BWRs complied with the recommendations of Bulletin 96-03 by installing larger, better designed, passive ECCS suction strainers. The design of these strainers considered the plant-specific suction strainer loadings of several types of materials, including LOCA-generated debris from dislodged thermal insulation and dislodged paint chips and rust accumulated in the suppression pool, which become thoroughly mixed in the suppression pool water by turbulence generated by the LOCA. In general, these loadings were much higher than predicted before the research following the Barsebäck event. These higher loadings increased the predicted flow resistance across the strainers that resulted in a decrease in calculated NPSHa. In some cases, this decrease prompted licensee requests to use containment accident pressure, which the staff approved after careful review.

In 1996 and 1997, because of conditions found in NRC inspections and reported in licensee notifications and licensee event reports, the NRC staff found that the NPSHa for ECCS and containment heat removal system pumps may not have been adequate in all cases (see, for example, NRC Information Notice 87-63, "Inadequate Net Positive Suction Head in Low Pressure Safety Systems," dated December 9, 1987). This finding applied to both PWRs and BWRs. To understand the extent of the problem, the NRC issued GL 97-04, "Assurance of Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997, which asked licensees to provide current information regarding NPSH analyses for the ECCS and containment heat removal pumps. In some cases, in response to GL 97-04, licensees revised their NPSH analyses, and in some of these cases, licensees proposed use of containment accident pressure in the calculation of NPSHa. Among the reasons for using containment accident pressure in determining NPSHa as a result of this generic letter were calculations that incorrectly omitted an important effect (such as underestimating flow losses), an increase in estimated debris loading on BWR ECCS suction strainers in response to NRC Bulletin 96-03, or an increase in suppression pool temperature (the result of degradation of the heat transfer capability of the heat exchangers in BWR suppression pool cooling systems).

The NRC reviewed all responses to GL 97-04. In some cases, especially those in which the licensee proposed use of containment accident pressure, the NRC performed detailed reviews. The NRC staff formulated and applied acceptance criteria for these reviews. A complete listing of these criteria was not documented in a publicly available source at that time. The applicable criteria were discussed in individual safety evaluations. To document these criteria for future use and to make them available to stakeholders, the NRC staff included them in Revision 3 of RG 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," issued November 2003. Regulatory positions on NPSH in this regulatory guide provide a single reference for most positions related to pump suction issues (vortexing, air entrainment, and debris blockage, as well as NPSH margin). GL 97-04 does not contain acceptance criteria; rather, it is a request for information. In particular, GL 97-04 contains no criterion prohibiting the use of containment accident pressure in the calculation of NPSHa.

The NRC has also approved use of containment accident pressure in calculating NPSHa as part of the approval of significant increases in reactor power (termed "extended power uprates"). These power increases result in increased decay heat that raises the temperature of the water being recirculated in BWR suppression pools or PWR sumps.

The NRC staff briefed ACRS several times on the general subject of the calculation and use of containment accident pressure in determining NPSHa. The staff also discussed the subject in the context of extended power uprates for two BWRs. Table 2 summarizes the NRC staff's interactions with ACRS on this topic from 1997 to the present.

Table 2 Staff/ACRS Interactions Regarding Use of Containment Accident Pressure

ACRS/ACRS SUBCOMMITTEE MEETING	MEETING DATE(S)	LETTER ISSUED (ACRS subcommittees do not issue letters).	COMMENT
ACRS 442nd Meeting	June 11–14, 1997	June 17, 1997	Briefing on NPSH related to GL 97-04. The ACRS letter states, "allowing some level of containment overpressure is not… acceptable…."

ACRS 447th Meeting	December 3, 1997	December 12, 1997 Agencywide Documents Access and Management System (ADAMS) Accession No. ML091130712)	Followup to June 1997 meeting. The ACRS letter agreed with use of containment accident pressure but suggested a broader perspective than just the LOCA.
ACRS 511th Meeting	September 11, 2003	September 30, 2003 (ADAMS Accession No. ML032731487)	Discussion of RG 1.82, Revision 3. Revision 3 contains NPSH guidance. This guidance was not discussed during the meeting.
ACRS Thermal Hydraulics Phenomena Subcommittee	July 19–20, 2005		Discussion of proposed revisions to RG 1.82, Revision 3, NPSH guidance.
ACRS 525th Meeting	September 8, 2005	September 20, 2005 (ADAMS Accession No. ML052630562)	Discussion of revisions to RG 1.82, Revision 3. Use of containment accident pressure should be "only selectively granted." RG revision should not be issued for public comment.
ACRS 526th Meeting	October 7, 2005	No letter	NRC staff informed ACRS of decision to risk-informed RG 1.82, Revision 3.
ACRS Power Uprates Subcommittee	November 15–16, 2005		Vermont Yankee Extended Power Uprate-Discussion of using containment accident pressure.
ACRS 528th Meeting	December 7, 2005	January 4, 2006 (ADAMS Accession No. ML060040431)	Vermont Yankee Extended Power Uprate-Discussion of using containment accident pressure. The ACRS letter states, "The ACRS has historically opposed a general approval of the use of containment overpressure credit." ACRS suggests a "more complete and rigorous consideration of uncertainties." Vermont Yankee extended power uprate received a favorable ACRS letter.

ACRS 539th Meeting	February 1, 2007	February 16, 2007 (ADAMS Accession No. ML070470314)	Browns Ferry Unit 1 5-Percent Power Uprate-Discussion of using containment accident pressure. Letter questioned acceptability of Appendix R scenario. Browns Ferry Unit 1 power uprate received a favorable ACRS letter.
ACRS 558th Meeting	December 4, 2008	No letter	NRC staff briefed ACRS on the draft guidance discussed in Section 6.0 of this enclosure.
ACRS 559th and 560th Meetings	February 5–7, 2009 March 5–7, 2009	March 18, 2009 (ADAMS Accession No. ML090700464)	Five conclusions and recommendations: containment overpressure should be "limited in amount and duration," any risk associated with operator actions to maintain adequate containment accident pressure should be acceptably small.
ACRS Subcommittee on Power Uprates	April 23, 2010		ACRS subcommittee briefed on staff draft guidance
ACRS 572nd Meeting	May 6, 2010	May 19, 2010 (ADAMS Accession No. ML101300332)	Staff briefed full committee on staff draft guidance. ACRS letter contains five recommendations.

In a December 12, 1997,[15] ACRS letter to the NRC Chairman, Shirley Ann Jackson, ACRS concurred with the NRC staff practice of using containment accident pressure in determining NPSHa, but urged the examination of all accident sequences (not just the LOCA). The ACRS letter was the result of a December 3, 1997, ACRS briefing by the NRC staff.

At an October 7, 2005, ACRS meeting, the NRC staff committed to risk-inform the regulatory guidance on using containment accident pressure in determining NPSHa. Revision 5 of SRP Section 6.2.2 (issued March 2007) introduces the use of risk insights into the regulatory guidance on this issue. It states the following:

> If containment accident pressure is credited in determining NPSHa, an evaluation of the contribution to plant risk from inadequate containment pressure should be made. One acceptable way of making this evaluation is to address the five key principles of risk-informed decision making stated in Section 2 of Regulatory Guide 1.174.

The NRC staff considers this guidance to be consistent with the recommendations of the December 12, 1997, letter from ACRS.

Positions 1.3.1.1 (for PWRs) and 2.1.1.1 (for BWRs) of RG 1.82, Revision 3, are consistent with the guidance of RG 1.1. Positions 1.3.1.2 (for PWRs) and 2.1.1.2 (for BWRs) explicitly allow use of containment accident pressure in determining NPSHa "where the design cannot be practicably altered."

Revision 5 of SRP Section 6.2.2 reflects the positions of RG 1.82, Revision 3, as it states that this regulatory guide "describes methods acceptable to the staff for evaluating NPSH margin."

As described above, guidance on containment accident pressure appears in several NRC documents. In addition to RG 1.1 and Revision 3 of RG 1.82, SRP Section 6.2.2, Revision 5, and the Office of Nuclear Reactor Regulation's (NRR's) "Review Standard for Extended Power Uprates," RS-001, Revision 0, dated December 24, 2003 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML033640024), contain guidance on containment accident pressure. The staff intends to revise these guidance documents to ensure that all these guidance documents are consistent with one another. At present, they are inconsistent. For example, RG 1.1 states that only the pressure prior to a postulated accident should be used while RG 1.82, Revision 3, permits an exception for operating reactors for which plant modifications are impracticable. Also, SRP Section 6.2.2 Revision 5 provides SRP acceptance criteria if containment accident pressure is credited in NPSH evaluations for containment spray pumps but does not mention the ECCS pumps. In addition, SRP Section 6.2.2 Revision 5 mentions risk analyses for containment accident pressure, but the other guidance documents do not.

Section 4.4 discusses containment accident pressure considerations related to Generic Safety Issue-191 (GSI-191). Section 4.6 discusses new reactor (active) designs.

4.3 Use of Containment Integrity or Containment Accident Pressure in Other Regulatory Contexts

Use of containment accident pressure (with the implicit assumption of containment integrity) is not unique to the determination of NPSHa. Several other important areas of safety analysis assume containment integrity.

Design-basis safety analyses, described in plant final safety analysis reports, assume containment integrity; that is, the containment is not assumed to leak at a rate greater than L_a. Appendix J to 10 CFR Part 50 defines L_a as the maximum allowable leakage rate at the peak calculated containment LOCA pressure, expressed as weight percent lost by leakage per 24 hours. Typical values are 0.1-percent for PWRs and 0.5-percent for BWRs.

Section I.D.2 of Appendix K to 10 CFR Part 50 permits the use of containment accident pressure in the calculation of peak cladding temperature. Predictions of core reflood following a large-break LOCA use containment accident pressure, although this pressure must be conservatively minimized (as required by Appendix K). Without this containment accident pressure, the peak cladding temperature criterion of 2,200 degrees F may be exceeded.

Radiological offsite dose analyses assume containment integrity (and thus the ability of the containment to retain accident pressure) by assuming that the containment leakage rate is equal to L_a.

In addition, an acceptance criterion for ATWS, the Appendix R fire, and station blackout is that the maintenance of containment integrity must be demonstrated.

The assumption of containment integrity is supported by the stringent requirements of the *Code of Federal Regulations* and the technical specifications as discussed in Section 5.1.

4.4 Generic Safety Issue 191 Considerations

Resolution of GSI-191 has resulted in the NRC issuing two generic communications dealing with the impact of LOCA-generated and latent debris on the ability of PWRs to recirculate water to the reactor vessel and core following a LOCA. NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors,” issued June 9, 2003, requested the status of compliance with regulatory requirements or a description of interim actions taken until the analysis was completed. NRC GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors,” dated September 13, 2004, requested an evaluation of the ECCS and containment spray system recirculation functions in light of the information on sump blockage by LOCA-generated and latent debris discussed in the generic letter. As a result of the generic letter, PWR licensees have modified their sump screen designs to accommodate the revised estimates of the amounts of potential debris. Although GSI-191 is not completely resolved for all operating PWRs, all have now installed larger sump screens or strainers.

The strainers were installed before testing determined the reasonably bounding head losses for each plant’s particular configuration and debris load. Additionally, the potential effects from chemical precipitates were not considered. Testing including chemical effects showed that head losses across the debris beds formed on the strainers could be considerably higher than those found without chemical effects.

As discussed in Section 1 of this enclosure, the NPSHa for many PWRs is determined with the assumption that the pressure at the liquid surface is equal to the vapor pressure of the sump water; that is, the sump water is saturated (the containment atmosphere and the sump water are at the same temperature and pressure). This approach ensures that partial pressure of the heated air above the liquid surface is not included in determining NPSHa, and NPSHa is therefore, conservatively underestimated.

Water in the sump will flash to steam in any location where pressure is reduced below the saturation pressure of the fluid. For the case of saturated liquid passing through a debris bed, flashing could occur as the result of the pressure drop that occurs due to flow across the debris bed. If the height of water above the strainer surface is greater than the pressure drop across the debris bed, flashing will not occur. Because sump strainers are often close to the surface of the sump pool and head losses can be relatively high, flashing can occur when it is assumed that the water is saturated, as assumed in NPSH calculations. Actually, the containment pressure is higher than the vapor pressure of the sump water because of the partial pressure of the heated air. This added pressure inhibits flashing as the fluid undergoes local pressure changes.

As part of the review of plant responses to GL 04-02, the staff requested that licensees evaluate whether flashing could occur across the strainers during recirculation.

The effects of fluid flashing in a debris bed have not been investigated experimentally as part of GSI-191. Flashing is a potential concern because flashing steam could cause an increase in head loss across the debris bed. Because the effects of flashing on strainer head loss are not understood, the staff has asked licensees to demonstrate that there is reasonable assurance that flashing will not occur across ECCS suction strainers. Many plants could not show that flashing would not occur under assumptions similar to those used for NPSHa evaluations (containment pressure equal to the vapor pressure of the sump water). These plants required the use of some pressure above the vapor pressure of the fluid to suppress flashing across the strainer. Because the plant and strainer designs could not be practicably changed to prevent flashing, the staff allowed licensees to perform conservative evaluations to show that flashing would not occur. These calculations assumed some credit for the partial pressure of the air in the containment atmosphere.

Some PWRs have designs that reduce the probability of flashing at the ECCS strainers. Plants with significantly subcooled maximum sump temperatures (those with ice condenser containments), larger strainer submergence, low strainer head losses, or vented sumps are less likely to experience flashing across their strainers.

PWR flashing calculations have been performed assuming conditions that increase the likelihood of flashing. The calculations use minimum strainer submergence, maximum sump water temperature, and maximum debris head loss. In some cases, the evaluations contain significant conservatism. The probability of these conditions coinciding is low.

Because the staff concluded that the flashing evaluations conducted by licensees were conservative and realized that there were limitations on how the strainers could be installed in existing containments, the staff accepted the use of limited containment pressure during its reviews of some licensee responses to GL 2004-02.

Loss of containment integrity (that is, loss of containment accident pressure), depending on the extent, before or during a LOCA would result in flashing of the heated sump water. This would reduce the sump water level and, in some cases, might reduce the water level below that needed for adequate sump screen performance. For this reason also, containment integrity (containment accident pressure) must be maintained during the LOCA.

4.5 Current Status of Plants Using Containment Accident Pressure

Among operating BWRs, 19 use containment accident pressure in determining NPSHa. All these BWRs have Mark I containments. The Mark I is the earliest BWR containment still in operation.

Out of 69 operating PWRs, 60 have sump water temperatures greater than 212 degrees F. Therefore, the sump liquid vapor pressure will be greater than the pressure in containment before the postulated accident. This vapor pressure is therefore, considered to be containment accident pressure.

Out of 69 PWR units, flashing is demonstrated not to occur for 31 of these units without crediting containment pressure above vapor pressure. The other 38 PWR units may require credit for containment pressure greater than saturation pressure and outside their current licensing bases to demonstrate that flashing will not occur under conservatively calculated conditions.

4.6 Use of Containment Accident Pressure in New Reactor Designs

In SRM SECY-06-0144, "Proposed Reorganization of the Office of Nuclear Reactor Regulation and Region II," dated July 21, 2006, the Commission directed the staff to consistently apply technical and regulatory standards, guides, and requirements both to new plant licensing and to operating plants and to look for other strategies, as appropriate, to achieve and maintain the desired consistency.

The only new reactor designs that have credited CAP to demonstrate adequate NPSH margin are the active PWR designs. Passive plants don't rely on active pumping systems, and the active BWR has not taken credit for containment pressure greater than the pressure before the postulated accident in NPSH analyses.

Active PWR designs for new reactors take a similar approach to the use of CAP in evaluating NPSH as operating PWR designs do. For example, both operating PWRs and new active PWR designs predict elevated sump water temperatures (greater than 100 degrees Celsius (212 degrees Fahrenheit)) and elevated pressures in containment during a design-basis accident. In calculating NPSH margin to demonstrate it is sufficient, new active PWR designs, similar to most operating PWRs, credit the inclusion of vapor pressure developed in containment during an accident in the NPSH margin calculation. (See Section 2.0 of Enclosure 1).

The regulation at 10 CFR 52.47, "Contents for applications; technical information," effectively requires design certification applicants to perform PRAs; an analogous regulatory requirement in 10 CFR 52.79 exists for combined license applicants. Therefore, the staff can request both design certification and combined license applicants to assess the risk significance of using CAP using the required PRA.

The staff will review the use of CAP in NPSH evaluations in the same manner for new reactors and operating plants, except for the treatment of risk.

5.0 TECHNICAL AND REGULATORY BASIS FOR ALLOWING USE OF CONTAINMENT ACCIDENT PRESSURE IN DETERMINING AVAILABLE NPSH

This section provides the technical justification for using containment accident pressure in determining NPSHa.

Licensees must demonstrate an acceptable level of safety when using containment accident pressure in determining NPSHa for design basis accidents and, for BWRs, those "special

events" which use CAP to provide adequate NPSH margin. The NRC staff considers this position to be consistent with ACRS letters to the Commission on this issue.[16,17]
The justification for using containment accident pressure in determining NPSHa is based on the following considerations:

- There is high confidence in the integrity of the containment to retain pressure.

- NPSH calculations for the design-basis LOCA are done conservatively. Water temperature is maximized, and containment pressure is minimized. NPSH calculations for beyond-design-basis events are done more realistically in accordance with NRC guidance for these events.

- The ECCS and containment spray pumps are of robust construction and of material resistant to cavitation damage.

- Emergency operating procedures either take into account reliance on containment pressure or are not significantly affected by reliance on containment pressure.

- The five key principles of risk-informed regulation are satisfied based on plant-specific information and analyses.

Each item is discussed in more detail below.

5.1 Containment Integrity

The original rationale for not using containment accident pressure in determining NPSHa, according to RG 1.1, is (1) the possibility of "impaired containment integrity" or (2) excessive operation of the heat removal systems (sprays or safety related fan coolers) resulting in a pressure less than that needed to maintain an adequate NPSH margin. The discussion of operator actions in Section 5.4 addresses the operation of containment sprays.

According to paragraph (o) in 10 CFR 50.54, "Conditions of Licenses," every primary containment for a water-cooled reactor shall be subject to the requirements in Appendix J to 10 CFR Part 50. In addition, 10 CFR 50.55a(2)viii and ix require detailed inservice inspections of concrete containments and their metal liners and metal containments, respectively, in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

Appendix J to 10 CFR Part 50 requires leakage rate testing of primary reactor containments. This includes determining the overall leakage rate of the containment (termed an integrated leakage rate test or ILRT or a Type A test) and the leakage rate of penetrations and containment isolation valves (termed a local leakage rate test or LLRT). The LLRTs of penetrations are termed Type B tests, and the LLRTs of containment isolation valves are termed Type C tests. Appendix J, Option B (the regulation now followed by all licensees), is a performance-based regulation; that is, the test intervals are established based on the previous performance (leakage rate) of the overall containment and its penetrations and isolation valves. Poor performance requires more frequent testing. Appendix J also requires that a visual examination of the accessible interior and exterior surfaces of the containment for structural deterioration be conducted before each ILRT and periodically in the time interval between ILRTs.

As stated above, for BWRs, only some Mark I containments use containment accident pressure in determining NPSHa. Containment integrity in a Mark I pressure suppression containment is continuously monitored during normal operation since the containment is inerted; that is, air is removed and the containment is filled with nitrogen gas. Any significant increase in the amount of nitrogen that must be supplied to the containment might be a sign of degradation in containment integrity and would be observed by the reactor operators. The reactor operators would then take the appropriate action in accordance with the plant abnormal operating procedures and technical specifications. Another sign of loss of integrity would be the presence of oxygen gas in the containment. Monitors provide continuous assurance that extremely low levels of oxygen are present. Again, if oxygen were detected, the operators would take the appropriate action in accordance with the abnormal operating procedures and technical specifications. Yet another indication of excessive leakage (loss of containment integrity), would be a change in the pressure difference between the drywell and the wetwell. The technical specifications define this pressure difference so as to limit certain hydrodynamic loads to acceptable values.

Several PWRs using containment accident pressure in determining NPSHa have subatmospheric containments. For PWRs with subatmospheric containments, containment integrity is, of course, also continuously monitored, and loss of integrity should be easily detected. For other containment types, such as large, dry PWR containments, containment integrity is monitored but in more subtle ways. The Electric Power Research Institute (EPRI) Report No. 1009325, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2, issued August 2007,[18] on extending containment leakage rate testing intervals provides the following description:

> …experience has shown that during normal reactor startup and during normal power operation it is fairly routine for most containment designs to either vent the overpressure that has built up (or to provide nitrogen for inerted containment designs) to maintain positive pressure within specified limits. The increase in pressure can be caused by increase in the average air temperature during heatup and startup, changes in barometric pressure, and an increase in the containment air mass from compressed air equipment bleeds and leakage. Absence or significant changes in the frequency of pressure buildup and venting over a substantial period will provide a qualitative indication of the existence of a containment atmosphere to outside atmosphere leak path. These factors, as well as others, provide additional means of detection of containment leakage pathways.

Large, dry PWR containments typically have a technical specification specifying upper and lower containment pressure limits. The corresponding surveillance requires the containment pressure to be checked typically every 12 hours.

In addition to pressure monitoring, typical features of containment isolation design and operation to ensure containment integrity include the following:

- The primary containment air lock(s) is a double door with limit switches on both doors that provide control room indication of door position. It is possible to open only one door at a time while the plant is operating.

- All power operated containment isolation valves have position indication in the control room.

- Primary containment isolation valves are controlled under plant procedures. Aspects include valve lineup checklists, locking of specific valves, second-party verification or independent verification of valve manipulations, and periodic surveillance of positions for accessible valves. The position of automatic isolation valves is indicated in the control room.

- Strict operability requirements apply to overall containment integrity and penetrations (air locks and isolation valves).

Containment integrity at accident pressure is assured by test. ILRTs must be performed at the peak calculated LOCA pressure. Some licensees conservatively perform the ILRT at the higher containment design pressure. At either pressure, it must be shown that the leakage rate does not exceed L_a.

Operating experience with regard to containment integrity is good. The EPRI report on extending the ILRT test interval mentioned above provides the results of a data survey of containment leakage rates based on ILRT data from 217 tests from reactors of different ages and containment designs. The data were obtained from 101 units. The conclusion of this survey is given below:

> Data from ILRT tests have been collected at various times to support various applications. In summary, two NEI [Nuclear Energy Institute] utility surveys and an examination of recent ILRT results provided ILRT data for 217 ILRT Type A tests that have been performed in the nuclear industry. Based on these data, the number of containment leakage events found during the performance of these tests is very small. In fact, no failures that would result in a large early release have been found. Leakage pathways detectable only by Type A ILRTs and therefore, influenced by the ILRT extension interval time have not been observed with magnitudes greater than 1.4 L_a.

Leakage rates of the order of magnitude of L_a, because of the small loss of mass from the containment atmosphere, do not significantly affect the containment pressure over the time of interest for using containment accident pressure for determining NPSHa in most cases.

The NRC and other organizations have tested scaled containments to failure to ascertain the integrity of containment structures and components under severe accident pressure loads. The following excerpt from the forward to NUREG/CR-6906, “Containment Integrity Research at Sandia National Laboratories, An Overview,” issued July 2006,[19] by the then-Director of the NRC Office of Nuclear Regulatory Research summarizes the results of this research:

> The results of the experiments and analyses summarized in this report clearly demonstrate that margins between the design capacity and ultimate capacity of containment structures are on the order of a factor of 2-5. The specific margin for a given plant is highly dependent on the details of construction…however, it is clear that the current design and construction methods ensure a margin of safety or capacity beyond the design condition.

Thus, it is highly probable that containment integrity is maintained even at containment loads greater than design pressure. Furthermore, these tests have shown that increased leakage does not occur until the pressure is significantly greater than the containment design pressure. The containment pressure when crediting CAP is much less than the containment design pressure.

The ILRT is performed at cold conditions because of the difficulty and impracticality of testing at accident temperatures. Experimental difficulties with the scaled containment tests have also prevented large-scale containment testing at elevated temperatures. However, NRC and EPRI experiments have been conducted for local structural details and penetrations. NUREG/CR-6906 reports that compression seals and gaskets, electrical penetration assemblies, and personnel airlocks were tested to DBA pressures and temperatures without failure.

A primary reactor containment typically has between 100 to 300 mechanical penetrations, which provide many potential leakage paths. Compression seals and gaskets in these penetrations represent an important part of the containment pressure boundary. These penetrations include equipment hatches, personnel air locks, and BWR drywell heads. Leakage behavior of those containment penetrations with elastomer seals is greatly influenced by the behavior of the elastomer seals. A paper by Parks and Clauss[20] summarizes NRC research on the performance of containment penetrations under severe accident conditions. If metal-to-metal contact does not exist between sealing surfaces of these penetrations, the gaskets represent the only barrier to prevent leakage through the containment at these locations. The following table from this paper summarizes the testing of compression seals.

Table 3 Summary of Compression Seal Test Data from Parks and Clauss

Material	Number of Tests	Test Environment	Range of Failure Temperatures (°F)	Mean Failure Temperatures (°F)
Ethylene Propylene	9	Steam	580–669	626
	8	Nitrogen	577–667	613
	1	Air	651	651
Silicone	8	Steam	486–592	512
	2	Nitrogen	>700	>700
	1	Air	681	681
Neoprene	3	Nitrogen	460–500	487

The test program includes unaged and aged test specimens (thermal only and radiation and thermal).

The test results indicated that the failure temperature was independent of applied aging, gap between sealing surfaces, gasket cross-section, or “out-of-flatness” of the sealing surfaces. The seals, made of all three typical materials (ethylene propylene, silicone, and neoprene), tested in three different test environments (nitrogen, air, and steam), failed at temperatures much higher than the design temperatures of light-water reactor containments. The tests did show that silicone has a lower failure rate temperature in steam than in air, while ethylene propylene was not similarly affected. These results are consistent with those of Hirao, et al.[21] However; the failure temperatures in steam are also greater than the design temperatures of light-water reactor containments. The BWR Mark I containment design temperature is 281 degrees F.

Brinson and Graves[22] point out some considerations in applying these test data to prototypical systems. In some cases, the flanges containing the seals are smaller than typical penetration seals and thus are more rigid. This makes the gap between flanges more uniform and less likely to move relative to each other at high temperatures. In addition, radiolytic damage may be sensitive not only to the total dose but the length of time over which the seals are exposed to this dose. These differences would be less significant under design-basis conditions than the severe accident conditions at which the tests were run.

The NRC has approved licensee requests to increase the interval between ILRTs, first from three tests in 10 years, then to one test every 10 years, and then to one test every 15 years. (The NRC approved the 15-year interval only on a one-time basis). Staff consultant calculations have shown an “imperceptible” increase in risk because of these changes in ILRT frequency.[23]

An important portion of the NRC staff Appendix R fire reviews deals with the spurious operation of equipment. This spurious operation is caused by hot shorts, open circuits, or shorts to ground in associated nonsafety circuits in fire areas. The staff review ensures that operation of safe-shutdown equipment is not prevented. For a plant using containment accident pressure in determining NPSHa, the staff Appendix R review will ensure that containment isolation is not adversely affected by spurious operation of equipment.

The testing and inspection requirements of Appendix J to 10 CFR Part 50 and 10 CFR 50.55a, “Codes and Standards,” the requirements of plant technical specifications, and plant procedures to ensure the correct positioning of containment isolation valves, as well as the testing of scaled containments, justify the assumption that containment integrity will be maintained at design-basis accident conditions.

Several instances of containment structure degradation have occurred over the past 25 years and have been addressed in a number of generic communications and reports. Most of the recent events have involved localized corrosion of the carbon steel liner of concrete containments, some with (liner) through-wall penetration, although with very small measured or calculated leakage potential. None of the events involved a loss of containment design function, including leak tightness assumed in the dose analyses. The NRC staff continues to monitor the industry response to these events, especially with regards to inspection methods and frequency.

5.2 Conservative Calculations and Statistical Approach

Current LOCA containment analyses done to determine NPSHa employ conservative assumptions that maximize the suppression pool temperature and minimize wetwell pressure. For instance, Table 4 lists some typical conservative assumptions made in calculating the LOCA conditions in BWR containment. Not every analysis includes each of these assumptions, but in general, containment LOCA analyses done for determining NPSHa are biased to overestimate the suppression pool temperature and underestimate the wetwell pressure. Although each of these assumptions biases the calculation in a conservative manner, they are not all of equal weight.

Table 4 Conservative Assumptions Used in BWR NPSHa Calculations

- The reactor is operating at 102 percent of licensed power.
- The worst single failure occurs.
- The wetwell air space is maximized.
- The initial drywell and suppression chamber pressures are at the minimum expected values to minimize the containment accident pressure.
- The maximum operating value of the drywell temperature and the maximum relative humidity (100 percent) are used to minimize the initial noncondensable gas mass and minimize the long-term containment pressure.
- The initial suppression pool temperature is the maximum technical specification value to maximize the calculated suppression pool temperature.
- Containment sprays are available to cool the containment. They are initiated at 600 seconds and operate continuously with no throttling of the RHR pumps (which provide the spray flow) below rated flow.
- Passive heat sinks are modeled to reduce containment pressure.
- All core spray and RHR pumps have 100 percent of the brake horsepower rating (rather than the water horsepower) converted to pump heat that is added to the suppression pool water.
- Heat transfer from the primary containment to the reactor building is neglected. This increases the calculated temperature of the suppression pool.
- Core decay heat is based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-5.1-1979, “Decay Heat Power in Light Water Reactors,” decay heat model with 2-sigma uncertainty.[24]
- Core decay heat is based on conditions bounding specific plant cycles.
- Feedwater flow into the vessel continues until all feedwater that would increase the suppression pool temperature has been added.
- The initial suppression pool water volume is the minimum allowed by the technical specifications to maximize the suppression pool temperature and minimize the positive contribution resulting from the static head.
- A single value of suppression pool level that is less than the calculated level at peak suppression pool temperature is used for the NPSH calculations.
- The RHR heat exchanger is assumed to have minimum effectiveness. Bounding values of tube plugging and tube fouling are included. Suppression pool temperature is sensitive to this parameter.
- Offsite power is lost at time = 0. This increases the actuation time of the ECCS.
- The service water (ultimate heat sink) temperature is at its technical specification or licensing-basis maximum value.
- Service water flow through the RHR heat exchanger is minimized. This further

	decreases the effectiveness of the RHR heat exchanger.
•	RHR flow, as it affects RHR heat exchanger performance, is minimized. This further decreases the effectiveness of the RHR heat exchanger.
•	For the LOCA, a conservative estimate is made of blockage of the suction strainers resulting from LOCA-generated debris. This increases the head loss (h_{loss}).
•	Break flow and containment spray flow are mixed with the containment atmosphere so that the torus pressure is minimized.
•	Containment leakage is equal to L_a.
•	Initial water level in the downcomers is minimized. This lowers the containment pressure.
•	The water flowing through the debris bed on the suction strainer is assumed to be at a temperature below the peak suppression pool temperature. This assumption results in a higher than expected head loss.
•	The temperature correction factor for NPSHr, which decreases the NPSHr as the temperature increases, is not applied.
•	The impeller characteristics of the most limiting pump are assumed to apply to all pumps of the same kind.

PWR containment calculations performed to determine NPSHa also use conservative assumptions, such as those in Table 5, to maximize the sump water temperature and minimize the containment pressure. Not every analysis employs every conservative assumption.

Table 5 Conservative Assumptions Used in PWR NPSHa Calculations

•	The reactor is operating at 102 percent of licensed power.
•	Core decay heat is based on ANSI/ANS 5.1-1979 with a 2-sigma uncertainty.
•	Offsite power is lost at time = 0. This maximizes the time to ECCS actuation.
•	Initial conditions are chosen to minimize containment pressure and maximize containment emergency sump water temperature.
•	The worst single failure occurs.
•	The worst break location is chosen (i.e., the break that gives the highest sump water temperature and the lowest containment pressure).
•	The pressure of the containment atmosphere is equal to the vapor pressure of the sump water at the sump water temperature. This results in the static height of water in the sump above pump suction elevation as the only positive contribution to the NPSHa.
•	The distribution of the energy released with the assumed break is distributed in the containment atmosphere in such a way that the sump temperature is maximized and the containment pressure is minimized. For older computer simulations, this is accomplished with the “pressure flash” model. More current, physically based analysis methods accomplish this by assuming that the liquid in the break flow that does not flash to vapor is dispersed as droplets of a size that maximizes the sump temperature and minimizes the containment pressure as the droplet falls to the containment floor and heat and mass are transferred between the droplets and the containment atmosphere.
•	All containment cooling systems (containment sprays and containment fan

coolers) are operating at design conditions to reduce containment pressure.

- A conservatively high value of initial containment temperature is assumed. This temperature could be based on containment coolers not in service before the postulated accident.
- The containment free volume is maximized.
- The RWST initial temperature is at its maximum technical specification value.
- The RWST level is at its minimum technical specification value. This causes recirculation to begin at a lower sump water level and with a hotter core.
- The sump recirculation switchover setpoint (RWST level) is at its maximum. This leaves more water in the RWST, which minimizes the water initially in the sump.
- The low-pressure injection and containment spray heat exchangers are at their minimum effectiveness (maximum aging effect and tube plugging).
- The service water (ultimate heat sink) temperature is at its technical specification or design value. (For example, the technical specification value could be 90 degrees F and the design value could be 95 degrees F).
- Flows through the shell and tube sides of the heat exchanger are reduced to minimize heat exchanger effectiveness.
- A conservatively long time is assumed for emergency service water flow to reach the low-pressure injection and containment spray heat exchangers.
- The temperature correction factor for NPSHr, which decreases the NPSHr as the temperature increases, is not applied.
- Sump water temperature away from the surface will be below the corresponding temperature at the surface because some heat will transfer out through the bottom of the containment and through piping on the way to the pumps. For example, one licensee calculated that 205 degrees F water cooled to 200 degrees F provides an additional 2.8 ft of NPSH margin. No credit is taken for this effect.
- The impeller characteristics of the most limiting pump are assumed to apply to all pumps of the same kind.
- The containment flood level is underestimated.
- The head loss resulting from debris on the sump screens is maximized.

Perhaps the most conservative assumption made is that all the conservative assumptions used in an analysis are assumed to occur <u>simultaneously</u>; that is, the break that yields the most adverse NPSH conditions occurs while the parameters specified in the technical specifications are all at their limiting values, the worst single failure occurs, and every physical process takes place in the most limiting way.

BWR calculations of NPSHa based on nominal containment conditions show that, for the LOCA, use of containment accident pressure is not required in some cases. The use of best-estimate values such as, for BWRs, the use of best-estimate heat transfer coefficients for RHR heat exchangers, use of nominal versus minimum suppression pool level, 100-percent initial reactor power rather than 102-percent, and use of nominal decay heat values rather than those at the 2-sigma upper tolerance limit make it unnecessary to use containment accident pressure in some cases.

A BWR calculation for one Mark I plant demonstrated that omitting the single-failure assumption (as part of a sensitivity study), with all other assumptions unchanged, and resulted in no need for the use of containment accident pressure.

Thus, it appears that a major contributor to the need for containment accident pressure for the LOCA NPSHa determination, at least in some cases, is excess conservatism.
Another approach to including uncertainty in the calculation of NPSHa is to calculate a nominal (best-estimate) value and add to this nominal value an upper tolerance limit. This upper tolerance limit would be determined by statistically combining the uncertainty for each significant input to this calculation. This approach has the advantage of defining and quantifying the degree of conservatism in the NPSHa value. The BWR Owners Group (BWROG) has proposed a similar approach,[25] which the staff has found acceptable. The safety evaluation has not yet been issued.

5.3 Pump Design

The pumps of interest are the RHR pumps and core spray pumps for BWRs and the low-head safety injection and containment spray pumps for PWRs.

These pumps are typically single stage and of robust construction. They are designed and manufactured to provide dependable service under relatively severe operating conditions (although the operating conditions are not as severe as in other process applications in which similar pumps perform). All pumps have mechanical seals and stainless steel impellers. The materials are chosen, in part, for their high resistance to erosion resulting from cavitation.[26] Marine condensate pumps with stainless steel first-stage impellers operate in continuous cavitation with satisfactory service life even though the temperature of the water is less than 100 degrees F (cavitation damage increases with decreasing temperature).

The various factors that affect the level of cavitation damage include the following:

- the suction energy or suction specific speed
- the impeller material
- the gas content of the pumped liquid

Suction specific speed and suction energy are characteristics of centrifugal pumps that indicate the tendency towards cavitation damage. BWR RHR pumps have a high suction specific speed. Hydraulic Institute Standard ANSI/HI 9.6.1-1998, “NPSH Margin,” states that “…generally speaking, high suction energy pumps are susceptible to noise and increased vibration, but will not suffer significant erosion damage (especially with more erosion resistant impeller materials).” Testing of ECCS pumps has shown that they are capable of cavitation operation for limited times with no damage or wear, as discussed below.

The data in Table 6 show the relative levels of cavitation damage for different materials used for pump impellers. Stainless steel is relatively resistant to erosion.

Table 6 Typical Cavitation Erosion Rate Data[27]

Material	Erosion Rate (milligrams/hour)
Mild Steel	50
Aluminum Bronze	5.6
Stainless Steel SS 316	11.3
Gun Metal (bronze)	34

NUREG-0897, Revision 1, provides guidance for correcting (increasing) the NPSHr as the concentration of air in the pumped liquid increases. The high-temperature water in a PWR sump or a BWR suppression pool during a DBA would not be expected to contain a significant quantity of gas (air or nitrogen).[28] In addition, a small concentration of gas in the water actually improves resistance to cavitation damage by adding some compressibility to the fluid to mitigate the shock of the imploding vapor.[29]

If cavitation were to occur during a postulated accident, it would occur with the pumped water at an elevated temperature. The potential for cavitation damage is less at elevated temperatures because of the reduced volume of vapor formed. For example, the steam/water specific volume ratio at 50 degrees F is 152,000, whereas at 212 degrees F (which is in the range of suppression pool maximum temperatures following a postulated accident), this ratio is 1,605, or about 100 times less. This volume difference affects the intensity of the shock resulting from cavitation bubble collapse.

ANSI/HI 1.1-1.5-1994, "American National Standard for Centrifugal Pumps for Nomenclature, Definitions, Applications and Operation,"[30] includes guidance in calculating a decrease in NPSHr with an increase in water temperature. This decrease in NPSHr can be significant. RG 1.82, Revision 3, does not recommend use of this generic temperature correction. According to the standard, it is based on "…operating experience in the field and a limited number of carefully controlled laboratory tests." It is also based on data from pumps of different types. The correction is also subject to caveats, such as no dissolved air or other noncondensables and no transient changes in temperature and absolute pressure.

In addition, because of the increased rate of change of the vapor pressure with temperature at higher temperatures, the uncertainty in NPSH resulting from uncertainty in temperature increases at higher temperatures. Thus, the apparent gain in a decreased NPSHr may be offset by the increased uncertainty in this value.

Positions 1.3.1.5 and 2.1.1.5 of RG 1.82, Revision 3, state that the temperature correction factor for NPSHr should not be used. Omitting this factor provides additional margin in NPSH calculations. This approach is consistent with the recommendation of a widely recognized expert on this subject.[31]

Licensees have tested cavitating RHR pumps for varying periods of time. These tests have not shown any material or mechanical damage. Table 7 summarizes cavitation experience with nuclear safety-related pumps.

Table 7 Results of Tests of Safety-Related Pumps in Cavitation

PLANT	PUMP TESTED	TEST SUMMARY AND RESULTS
Beaver Valley 1[32]	Recirculation Spray	NPSHa lowered below NPSHr by water level and temperature in a tank in a closed loop. Pump run for ½ hour at 2,000 gpm with NPSHa = 5 ft. NPSHr approximately 6 ft. Pump disassembled and examined. No visible signs of wear or cavitation.
Beaver Valley 2[33]	Low-Head Safety Injection	NPSHa lowered below NPSHr by water level and temperature in a tank in a closed loop. No damage found.
Quad Cities/Dresden[34]	RHR	Closed-loop cavitation test at capacities of 4,000, 5,350, and 6,000 gpm. Pump disassembled. No evidence of damage. Pump reassembled for run-out cavitation test at 6,000 gpm. Suction pressure reduced until below cavitating point of previous test. Condition maintained for 1 hour. At completion of test, impeller removed and inspected. No damage. Pump reassembled for second run-out cavitation test at 6,000 gpm, and NPSH value slightly below the guaranteed value (approximately 40.2 ft). Suction valve closed further. Flow fell to 5,700 gpm. Maintained this flow for 1 hour. Impeller, shaft sleeve, and bearing in excellent condition except for minor scratches.

Browns Ferry 3[35, 36]	RHR	Tests of Browns Ferry RHR pumps in cavitation and at run-out conditions demonstrated acceptable pump performance while operating in regions beyond that certified by the pump manufacturer. The pumps were found to operate acceptably when the NPSHa was degraded to about 60 percent of the manufacturer's limit for flow rates in the range of 8,000 to 10,000 gpm. For flows in the range of 12,000 gpm, the pumps operated successfully at 73 percent of the manufacturer's specified limit. The tests found 10-percent degradation in discharge head, acceptable pump vibration, but severe cavitation noise. The tests were not carried out to the value of NPSH at which there would be a sudden severe loss of pump discharge head. The Browns Ferry pumps are typical of those used in BWR Mark I containment plants (vertical, deep-well centrifugal pumps). The Browns Ferry tests also investigated the effect of throttling flow. With the flow throttled to 80 percent of nominal flow, the pumps performed well down to 54 percent of the manufacturer's specified limit.
Crystal River Unit 3[37]	Containment Spray Pump	Pump vendor expressed "no reservations" in pump operation with NPSHr defined as 5-percent breakdown in total dynamic head (TDH) rather than the standard 3-percent value. An example of this gain in margin from the reduction in NPSHr is given below for a containment spray pump. 3-percent breakdown in TDH[+] @ 1,359 gpm: 14.8 ft 5-percent breakdown in TDH @ 1,359 gpm: 12.5 ft

Vermont Yankee[38]	RHR Core Spray	The pump vendor provided recommendations on operation with reduced NPSHr for both the RHR and core spray pumps based on cavitation tests on the same and similar pumps. RHR Pumps: 7 hours at an NPSHr above the 6-percent head drop followed by a ramp in the NPSHr value to a value judged acceptable for 8,000 hours of operation. Core Spray Pumps: 7 hours of operation at less than 3-percent limit followed by a ramp in the NPSH value to a value judged acceptable for 8,000 hours of operation.

The staff has accepted predictions of operation in cavitation for a limited time based on testing under cavitation conditions for the time period that the pump is predicted to cavitate and when post test inspections verify that no damage resulted from such operation. This is consistent with the guidance of RG 1.82, Revision 3. Draft guidance developed by the staff, described in Section 6.0 of this paper, provides improved guidance on testing under cavitation conditions.

5.4 Impact on Emergency Operating Procedures

BWR emergency operating procedures have cautions and restrictions associated with NPSH.

Operators would have indications in the control room should ECCS pump cavitation occur. Typical indications would include the following:

- erratic or decreasing pump motor current,
- erratic flow or flow less than expected, and
- frequent adjustments to the ECCS discharge valve to maintain a constant flow rate.

In response to these indications, the operators, in accordance with operating procedures, will take the following actions:

- Throttle (reduce) the pump flow.

- Remove a pump from service if not needed to restore or maintain emergency operating procedure parameters.

- Realign the suction source from the suppression pool to another source and replenish that source.

- Consider using other systems and water sources, such as the service water or the fire protection system.

Calculations have shown that throttling the flow is an effective way to increase NPSHa by reducing the head losses in the system while maintaining cooling and makeup flow. NPSHr would also be reduced, thereby increasing NPSH margin.

PWRs could take preemptive steps, such as terminating a train of containment spray, to prolong use of the RWST. Some licensees are considering eliminating automatic actuation of containment spray following a LOCA. The staff has approved such a request for Fort Calhoun Station Unit 1.[39]

Licensees mentioned the option of refilling the RWST in response to NRC Bulletin 2003-01. PWR ECCS pumps usually cannot be throttled, but are set in a fixed configuration to ensure flow balancing. It may be possible, as discussed previously, for the pump to perform acceptably while cavitating. A study looking at the risk reduction for a PWR found that such actions, taken when the sump is blocked or partially blocked by debris generated by DBAs, reduced the core damage frequency (CDF) “substantially.”[40]

As stated earlier, the NRC based its position that containment accident pressure should not be used in determining NPSHa, as stated in RG 1.1, on potential loss of containment integrity and overcooling of the containment atmosphere. Section 5.1 of this report discussed containment integrity.

Operation of safety-related containment cooling systems (sprays and fan coolers) will reduce the temperature, and therefore, the pressure of the containment atmosphere.

Emergency operating procedures for plants using containment accident pressure would need to provide guidance to the operator on the amount of containment pressure that must be maintained. BWR emergency operating procedures contain curves of pump flow rate versus suppression pool temperature with containment pressure as a parameter. For a given pump flow rate and suppression pool temperature, the containment pressure must be greater than the appropriate curve to maintain adequate NPSHa. This provides unambiguous indication to the operator that adequate NPSH margin exists.

In addition, BWR NPSH calculations assume that drywell and wetwell sprays operate for the duration of the accident (after initiation at 10 minutes). For PWRs, similar procedure and analysis considerations would be used. Therefore, loss of sufficient containment accident pressure as the result of overcooling is not a significant concern.

5.5 Risk Considerations

As previously discussed, the determination of adequate NPSH margin is based on a deterministic analysis. The licensee examines the response to the design-basis LOCA. For BWRs, the licensee also examines other events that result in raising the temperature of the suppression pool. These include the Appendix R fire, station blackout (after the coping period when the suppression pool must be cooled), and anticipated transients without scram (ATWS) events.

There are several supporting requirements in the American Society of Mechanical Engineers PRA standard[41] that relate to the use of containment accident pressure to provide adequate NPSHa to the ECCS pumps. Supporting Requirement AS-B2 requires PRA developers to include the impact of system dependencies in either the accident sequence models or in the PRA system models, and specifically mentions containment heat removal in this context. Supporting Requirement AS-B3 requires PRA developers to identify the phenomenological conditions created by the accident progression that could impact the success of the system or function under consideration, specifically mentioning the loss of NPSH margin.

The risk implications of crediting containment accident pressure to provide adequate NPSHa to the ECCS pumps have been assessed in conjunction with two recent BWR extended power uprate applications. These evaluations indicate that the increase in the internal events core damage frequency is less than 10^{-6} per year. Neither of these evaluations provided a quantitative assessment of the impact on core damage frequency from external events (e.g., seismic and internal fires).

The NRC Office of Nuclear Regulatory Research (RES) has investigated the risk of crediting containment accident pressure for a BWR/3 with a Mark I containment. This analysis was conducted to estimate the increase in CDF that results from relying upon containment accident pressure (CAP) to prevent ECCS pump cavitation. The analysis was limited to the study of all internal initiating events that are currently contained in the Standardized Plant Analysis Risk (SPAR) models (transients and LOCAs). External events were excluded due to lack of detailed cable routing and seismic fragility information. The leak size needed to prevent adequate NPSH margin is plant-specific and is determined through containment thermal-hydraulic analyses. For the plant used in the RES evaluation, the analysis assumed 20 L_a. Three timeframes were considered: 1) pre-initiator – containment may have a leakage path before an initiating event; 2) upon-initiator – containment may fail to isolate when an initiating event occurs and 3) post-initiator – containment may start to leak after the initiating event occurs.

The RES evaluation found that the large LOCA is the only initiating event where the loss of containment integrity leads directly to core damage. These scenarios are dominated by pre-initiator failures of containment. The probability of a pre-initiator containment leak depends on how the containment integrity is determined. This may be done by tracking the amount of nitrogen needed to maintain the containment inerted or oxygen concentration monitoring. Pre-existing leakage probabilities were determined to be sensitive to the interval during which a leak may go undetected (“non-detection interval”) and not so sensitive to other parameters. The relationship between this detection interval and pre-existing leakage probability is fairly linear when the leak could be detected monthly or more frequently. The change in CDF due to credit for containment accident pressure for the RES-analyzed plant was found to be very small ($<10^{-6}$/yr, as defined in RG 1.174) when the non-detection interval is one month. This is considered a bounding non-detection interval for an inerted containment.

Based on the above analyses and results, if a licensee submitted a risk-informed application to use containment accident pressure for determining the available NPSHa for ECCS and containment heat removal pumps, the staff believes the application would meet the risk acceptance guidelines of Regulatory Guide 1.174 and would be justified when complemented by the deterministic assessments described in Section 6.0.

6.0 STAFF GUIDANCE

6.1 Purpose of Guidance

The NRC staff (staff) has developed draft guidance for the use of containment accident pressure. The staff presented this guidance to ACRS on May 6, 2010. The staff transmitted this draft guidance to the BWROG on March 1, 2010 (ADAMS Accession No. ML100550903) and the Pressurized-Water Reactor Owners Group on March 24, 2010 (ADAMS Accession No. ML100740516). The staff has requested information from the BWROG on the uncertainties in various factors contributing to the total uncertainty in NPSHr and the behavior of pumps in cavitation based on performance history and design.

The purpose of the guidance being developed by the NRC staff is to ensure that ECCS and containment heat removal pumps will perform their safety functions during postulated DBAs and certain postulated non-DBAs. This is accomplished by focusing on the capability of the pumps to perform their safety functions during postulated accidents. In particular, based on recommendations by ACRS, the guidance quantifies both uncertainty and margin. Past industry practice, accepted by the NRC staff, has been to perform conservative (i.e., bounding) analyses for the DBA (e.g., the LOCA), and realistic analyses for the non-DBAs. While the conservative analyses produce bounding results, they do not provide a measure of the uncertainty or of the margin to a more expected result.

The guidance in this paper is applicable to both BWRs and PWRs. However, this paper emphasizes BWRs because the more recent applications has been to BWRs.

6.2 Approach to Uncertainty

The current approach to calculating NPSH margin assigns bounding values to the parameters used in the calculation of NPSH margin. These bounding values and assumptions are typically based on historically high or low values or on technical specifications limiting conditions for operation. The chosen accident scenario is also limiting. For example, the worst pipe break (giving the most limiting NPSH margin) is assumed for the LOCA, and the worst single failure is assumed. It is also assumed that all these limiting conditions occur simultaneously.

For the DBA consisting of a LOCA, the BWROG has proposed[42] an alternate method to the conservative method of calculating the NPSH margin which uses a Monte Carlo calculation to determine the containment accident pressure. Input values for some parameters are sampled from statistical distributions, and conservative (bounding) values are used for others. An acceptance criterion of 95-percent probability at a 95-percent confidence level (95/95) is used for the Monte Carlo pressure calculation. However, since conservative values are used for other input to the NPSHa calculation, the tolerance limit on NPSHa is actually greater than the 95/95 value.

For the non-design-basis events, the NRC guidance allows more realistic input values to be used. In addition, the assumption of a worst single failure is not applicable.

In accordance with this guidance, licensees should attempt to quantify uncertainty in the calculation of NPSH margin whenever possible. If this is not possible, the use of bounding or historically high or low values is recommended.

6.3 Guidance on the Use of Containment Accident Pressure in Determining the NPSH Margin of Emergency Core Cooling System and Containment Heat Removal System Pumps

6.3.1 Required NPSH and Effective Required NPSH ($NPSHr_{eff}$)

NPSHr is a property of the pump itself. In addition to the pump design, NPSHr varies with the pump flow rate and the temperature of the pumped water.

The NPSHr corresponds to an acceptable level of pump cavitation; that is, the pump will accomplish its safety function with that level of cavitation for the time necessary. The necessary amount of time should include not only the duration of the accident when the NPSH margin may be limited, but any additional time needed for operation of the pump after recovery from the accident when the pump is needed to maintain the reactor or containment, or both, in a stable, cool condition but at a much greater NPSH margin. This additional time is usually taken to be 30 days.

For practical application, the Hydraulic Institute has defined NPSHr as corresponding to a decrease in pump total dynamic head (TDH) (see Section 2.0 of this enclosure) of 3 percent.[43] This value of NPSHr will be denoted $NPSH_{3\%}$. Although this definition is useful, it does not correspond to a physical process in the pump. At this value of NPSHr, significant pump cavitation has already occurred. It can take from 1.05 to 2.5 times $NPSHr_{3\%}$ to achieve the 100-percent head point, and typically four to five times $NPSHr_{3\%}$ to totally eliminate cavitation.[44] This ratio can reach 20 for very high suction energy pumps.

Two values of NPSH do correspond to physical limits. The first is the value of NPSH corresponding to the inception of cavitation, $NPSH_i$. Determining cavitation inception (i.e., the first appearance of vapor) requires either visual observation or careful testing with sophisticated instrumentation. The second value of NPSH corresponds to a complete breakdown of the flow; that is, there is a total collapse of pump head. Since neither of these limits is of practical use, the Hydraulic Institute has chosen $NPSHr_{3\%}$ because it is relatively easy to determine.

The staff proposes that the NPSH margin be calculated from NPSHa − $NPSHr_{eff}$, where $NPSHr_{eff}$ is the $NPSHr_{3\%}$ value with uncertainties in NPSHr included. This calculated NPSH margin should be equal to or greater than zero.

NPSHr, as a function of flow rate, is typically obtained by testing the pump in question or a similar pump at the pump vendor's facility in accordance with the Hydraulic Institute standard[45]. The test begins with a large value of NPSHa in the test loop, which is gradually reduced. The flow rate and the pump speed are held constant. As the test loop NPSHa is reduced, a value of NPSHa is reached at which the pump TDH can no longer be maintained and decreases. The value of NPSHr is the value of the measured NPSHa for the test corresponding to a given measured decrease in the TDH. The Hydraulic Institute defines NPSHr as corresponding to a TDH which is 3 percent below the TDH at higher values of NPSHa for which the TDH is constant. This is $NPSHr_{3\%}$. Other values could also be obtained (e.g., $NPSHr_{1\%}$). These test methods are the most accurate for determining the $NPSHr_{3\%}$ of a pump. For best accuracy, the test should be conducted at the rated speed and impeller diameter, with the NPSHa controlled by a vacuum pump.

The resulting net NPSHr accuracy of this method would be expected to be in the range of ± 1 to 2 ft or ± 2.5 percent to ± 5 percent, whichever is larger, depending on the accuracy of the instrumentation and air content of the test liquid.[46] However, experience has shown that the uncertainty in NPSHr of a pump installed in the field is greater than the uncertainty obtained by testing at the pump vendor's facility for the following reasons:

(1) The NPSHr varies with changes in pump speed caused by motor slip.

(2) The NPSHr decreases with increasing water temperature.

(3) Incorrectly designed field suction piping adversely affects the NPSHr.

(4) The air content of the water used in the vendor's test may be lower than that of the pumped water in the field.

(5) Wear ring leakage impacts NPSHr.

The following paragraphs explain these reasons in greater detail.

<u>Pump speed</u>. The NPSHr varies as the square of the pump speed, which changes with changes in the motor slip. Operation at less than full-rated motor power or with high-efficiency motors tends to reduce motor slip. Motor slip can cause the pump to operate at slightly higher speeds in the field compared to a factory test speed with the factory calibrated motor.

<u>Water temperature</u>. The NPSHr decreases with increasing water temperature. This increases NPSH margin. Pump vendor tests are run with cold water. As the water temperature increases, the specific volume of the vapor decreases, thus creating less void blockage. The enthalpy for creating vapor also decreases as the temperature increases. The American National Standard for Centrifugal Pumps for Design and Application, ANSI/HI 1.3-2000[47] provides curves to adjust the NPSHr for higher temperatures, lowering the value of the estimated NPSHr.

However, the effect of the uncertainty in the calculated water temperature should be taken into account. If the calculated water temperature is greater than the expected value, the magnitude of the reduction in NPSHr, which results in more apparent margin between NPSHa and NPSHr, could be offset by a decrease in margin resulting from the effect of an increase in the vapor pressure on NPSHa which would reduce the apparent margin.

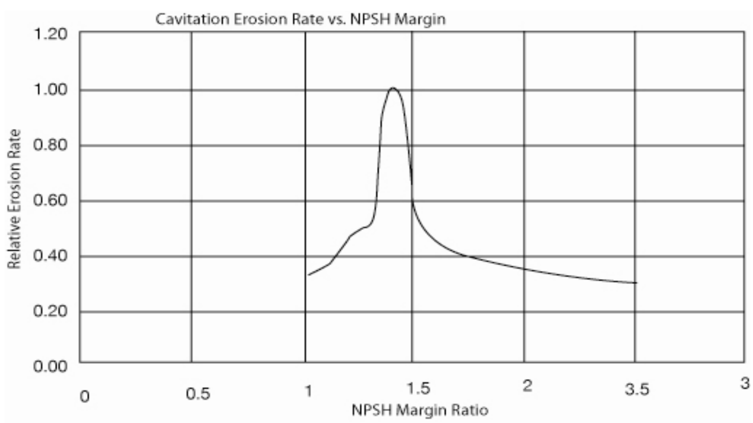
Therefore, this effect should not be included in determining the NPSHr at higher water temperatures.

<u>Incorrectly designed suction piping</u>. An approach flow as uniform as possible and free of swirl (prerotation of the water before entering the impeller) and vortices is important for acceptable pump operation. Suction piping should be short and straight. This is not always possible in field configurations. The pressure drop in the piping should be minimized to obtain the maximum NPSHa. High and very-high suction energy pumps (such as the ECCS and containment spray pumps) are more susceptible to problems from poor suction piping conditions. <u>The air content of the water used in the vendor's test may be lower than that of the pumped water in the field.</u> Another factor which affects the NPSH margin is the release of noncondensable gases (such as air or nitrogen) dissolved in the water as the minimum pressure in the pump approaches the

water vapor pressure. The air produces several effects: (1) air dampens the effect of cavitation by lessening the effect of the shock from implosion of the condensing vapor bubbles, which causes the cavitation erosion damage, (2) air in the pumped water also increases NPSHr, and (3) air may interfere with the water cooling of pump seals.

The solubility of air or nitrogen in water decreases with increasing temperature. This decreases the gas entrained in the pump flow.

The figure below illustrates the effect of air coming out of solution on the erosion rate. The figure is a plot of cavitation noise (from bubble collapse)-measured with acoustic instrumentation-as a function of the NPSH margin ratio (NPSHa/NPSHr). The cavitation noise is a measure of the intensity of cavitation occurring in the pump and correlates with the extent of cavitation erosion. The cavitation noise reaches a maximum value at an NPSH margin ratio greater than 1.0 (NPSHa = NPSHr) and decreases. The decrease results from the effect of air coming out of solution with the vapor formation and cushioning the effect of the vapor bubble collapse. The cavitation acoustic noise (an indication of the cavitation erosion rate) is greatly reduced as the amount of air coming out of solution is increased. Budris and Mayleben[48] describe tests that demonstrate this effect.



Typical Relative Erosion Rate vs. NPSH Margin near BEP Flow Rate

The second effect of noncondensable gas coming out of solution at the low-pressure region within the pump is to increase NPSHr by creating additional blockage. Sufficient gas can interrupt pumping altogether by “gas locking” or “gas binding.” Both entrained and dissolved air and gas will increase the NPSHr of a pump because of the added blockage of entrained and dissolved air at the low local internal pressures within the pump. (See Figures 2 and 4-2 of NUREG/CR-2792, “An Assessment of Residual Heat Removal and Containment Spray Pump Performance under Air and Debris Ingesting Conditions,” issued September 1982.[49])

NUREG/CR-2792 discusses the effects of air and proposes an “arbitrary” relationship between NPSHr and the fraction of air at the pump suction. The relationship proposed in NUREG/CR-2792 is:

$$NPSHr_{air/water} = NPSHr_{water} (1 + 0.5\ AF)$$

where AF is the air volume fraction in percent. As shown in Figure 4-3 of NUREG/CR-2792, this relationship significantly overpredicts the effect of air on NPSHr for the one set of data presented.

Penney[50] proposed a method of determining the effect of noncondensable gas coming out of solution on the NPSH margin. Tsai[51] and Chen[52] expanded upon this method. Penney's method calculates the pressure at the pump eye necessary to ensure that no more than the acceptable void fraction of noncondensable gas comes out of solution. A paper by Wood, et al.[53] provides an example of a successful application of this method.

The Penney method assumes that cavitation starts in a centrifugal pump at NPSHr. As stated earlier, cavitation actually starts at from 2 to 20 times the NPSHr value. The method also assumes that the pressure at the inlet to the pump impeller is uniform, and the entire suction area of the pump reaches the vapor pressure at the same time. The flow through the impeller is actually very complex, with local pressure depressions caused by locally high velocities. Therefore, not all of the inlet flow cavitation liberates dissolved gas at the same time. In addition, there is a time delay to the release of dissolved gas in a liquid. Not all of the dissolved gas comes out of solution at the same time.

Therefore, the staff concludes that the Penney method is overly conservative. The effect of air on NPSHr can be included as an uncertainty component of $NPSHr_{eff}$.

Section 6.3.5 of this report discusses the mechanical effects of air on pump operation.

Wear ring leakage. This is an important factor that can have a marked impact on pump NPSHr because of the added flow and disturbance from increasing the clearance of the pump wear rings. The effect can be to increase NPSHr on the order of 20 to 50 percent when the wear ring clearance is doubled, which is the typical point at which wear rings are replaced. The actual impact of wear ring leakage is determined by the amount of wear, the flow rate (as a percent of best efficiency flow), and the specific speed of the pump. The NPSHr of a high-specific-speed pump will be affected less by wear ring wear (since the leakage is a smaller percentage of the net pump flow rate), compared to a low-specific-speed pump, which has a lower flow rate and higher pressure which causes more leakage. Finally, the NPSHr impact is normally greatest at the highest flow rates.

Since this uncertainty is a result of pump wear, emergency pumps that are only operated infrequently should not need this NPSHr correction, since they should experience little wear. However, for LOCA pumps that do experience substantial operation, a pump wear uncertainty in the range of +0 to 50 percent should be considered, with 40 percent being a likely reasonable default value. Licensees should determine a value of $NPSHr_{eff}$ applicable to their pumps, taking into consideration the effect of suction piping, air content of the water, motor slip, and wear ring leakage.

For nondesign-basis events, such as the BWR events that result in raising the temperature of the suppression pool (e.g., shutdown after an Appendix R fire, an ATWS, or station blackout), no uncertainty on $NPSHr_{3\%}$ is required. NPSHa may also be calculated using realistic (rather than conservative) assumptions for these events.

6.3.2 Available NPSH Less Than Effective Required NPSH

It is possible that the NPSHa may be less than $NPSHr_{eff}$. In this case, allowing operation in this mode is acceptable if the results of tests in which the pump is run in cavitation and inspected after the test show acceptable wear and no damage. The following conditions should apply:[54]

- Predicted operation during the postulated accident below $NPSHr_{eff}$ (LOCA) or $NPSHr_{3\%}$ (nondesign-basis event) is of limited duration (less than 100 hours).
- The tests are conducted on the actual pump with the same mechanical shaft seal (including flush system) or at least a pump of the same model, size, impeller diameter, materials of construction, and pump seal and flush system.
- The test is conducted at the same (field application) speed.
- The test is conducted at the actual predicted NPSHa since testing at a lower NPSHa can actually reduce, rather than increase, the cavitation erosion rate in some cases.
- The test duration should be for the time NPSHa is predicted to be less than $NPSHr_{eff}$ (LOCA) or $NPSHr_{3\%}$ (non-design-basis event).
- The flow rate and discharge head must remain above the values necessary to provide adequate core and containment cooling.

6.3.3 Cavitation Erosion and the Use of Containment Accident Pressure

One of the adverse effects of insufficient NPSH margin is cavitation, which results in erosion (pitting) of the surface of the impeller blades and possibly other parts of the pump from the condensation (implosion) of vapor bubbles near a solid surface.

Visual studies, acoustical measurements, and field experience show that the region of maximum cavitation erosion rate occurs at an NPSHa value between the $NPSHr_{3\%}$ value (NPSH margin ratio = 1.0) and the point of cavitation inception (NPSH margin ratio of 4.0 or higher). The exact value will vary depending on the pump, the amount of air dissolved in the water, and the point of operation on the pump curve with respect the best efficiency point.

The NPSH curve of incipient cavitation and the NPSH curve of maximum cavitation erosion have the same shape. Both curves peak at the run out capacity. Both curves then decrease to the point of “shockless entry.” At this point the impeller pressure distribution is most favorable, and incipient cavitation and cavitation erosion are both minimized. As the flow rate further decreases and incipient pump cavitation occurs at higher values of NPSH, the NPSH corresponding to maximum erosion rate also increases. Hydraulic instabilities may occur in this region.

Since the NPSHa depends on the containment pressure, which the operator cannot control (except to limit it through the use of containment sprays or fan coolers, or both), the NPSHa will vary during a postulated accident and could spend some time in the region of the maximum erosion rate.

Pump tests indicate that the zone of maximum erosion rate lies between NPSH margin ratios of 1.2 to 1.6 for pumps operating outside of the zone of suction recirculation.[55,56] While very high suction energy pumps, such as BWR RHR and core spray pumps, are subject to cavitation erosion, the time operating in the maximum erosion zone has not been correlated with the degree of damage. Therefore, an open issue is the length of time a pump may operate in the maximum cavitation zone without failing and how this cumulative time to failure relates to the pump mission time. The staff is soliciting additional information and data to better define the need for and length of a time limit. For this paper, the staff selected a time limit of 100 hours for the time permitted in the maximum erosion zone.

6.3.4 Containment Accident Pressure and Available NPSH

In addition to consideration of the NPSHr and the adverse mechanical effects of cavitation, it is necessary to determine the NPSHa to determine the NPSH margin.

If the calculated NPSHa, assuming the containment pressure is at its pre-accident value, is less than $NPSHr_{eff}$, the containment pressure is increased so that NPSHa equals $NPSHr_{eff}$. The increase in containment pressure necessary for NPSHa to equal $NPSHr_{eff}$ is the amount of containment accident pressure used. The containment accident pressure used must be less than the total containment accident pressure at that time.

To determine the NPSHa, the temperature of the pumped water and the pressure above the water free surface in the suppression pool or sump must be known. The flow losses in the suction piping from the water source (suppression pool in a BWR or sump in a PWR) must also be known.

The containment accident pressure, water temperature, and water elevation above the pump suction should be calculated with an NRC-approved method. Calculation of containment accident pressure and water temperature involves heat and mass transfer processes within the containment and the tracking of the water, gas, and vapor inventory in the containment. The modeling of plant equipment (e.g., flow rates) is also necessary. The mass and energy flows into the containment caused by the postulated accident must also be determined.

The containment calculations for NPSH analyses are typically conservative. In this approach, all parameters that have a significant effect on the containment pressure and water temperature are assumed to be simultaneously at bounding values; these values are typically either technical specification limits, such as limiting conditions for operation, or values known to bound the expected value of a parameter.

The BWROG has proposed[57] a Monte Carlo method for calculating the lower tolerance limit (e.g., the 95/95 value) of the variable H_{ww}, which the BWROG topical report defines in the following manner:

$$H_{ww} = \frac{P_{ww} - P_v}{\rho g}$$

where

P_{ww} = the wetwell pressure above the pool surface

P_v = the water vapor pressure

ρ = the water density

g = gravity acceleration

H_{ww} consists of two of the terms in the equation for the NPSHa obtained from the containment analysis. The other two terms in the calculation of NPSHa are the elevation of the water level above the pump suction centerline and the flow losses. The elevation of the water level above the pump centerline can also be determined in the containment calculation, or a conservatively low value may be used. The flow loss term in the NPSH equation is calculated conservatively to bound the expected value. For the LOCA, the flow loss term includes the flow resistance caused by the accumulation of debris on the suction strainers or screens upstream of the pump suction.

The NRC staff performed independent Monte Carlo, conservative, and more realistic calculations for a LOCA in a typical BWR/3 with a Mark I containment. A staff memorandum documents these calculations.[58] Portions of the phenomena identification and ranking table (PIRT) method[59] were applied to the determination of NPSH margin to identify the important parameters to be considered. Sensitivity studies were performed for a BWR/3 with a Mark I containment using the GOTHIC computer code[60] for those parameters determined to be significant. In addition the single-failure criterion was assumed to apply in both the conservative and the Monte Carlo calculations. The containment (drywell and wetwell) sprays were assumed to be in operation after 10 minutes for the duration of the event. The analyses assume no operator action before 10 minutes.

Variables that tend to increase the suppression pool (or sump) temperature have the greatest effect in minimizing the NPSHa. These variables include the reactor power, decay heat, initial suppression pool temperature, RHR heat exchanger effectiveness, and service water temperature. The other parameters in the table affect the containment pressure. They are chosen so as to minimize the drywell and wetwell pressures.

The BWROG methods did not include computer code modeling uncertaintly. The staff finds this acceptable since the BWROG method uses the General Electric Hitachi computer code, SHEX, which is biased conservatively. The staff calculations using GOTHIC have verified this conservatism.

Staff calculations in this paper use the GOTHIC computer code. No modeling uncertainty for the GOTHIC code is publicly available. GOTHIC predictions of Marviken (a Swedish BWR with a vent system similar to U. S. Mark II BWR containment) blowdown test data for wetwell pressure, drywell-to-wetwell differential pressure, and wetwell liquid and vapor pressure temperatures show good agreement between the GOTHIC code and the Marviken data. Licensees proposing to use containment accident pressure in determining NPSHa should also perform a realistic calculation of NPSHa to compare with the conservative calculation or the Monte Carlo 95/95 calculation.

6.3.5 Effect of Noncondensable Gas on Pump Mechanical Performance

The amount of entrained air in a pump increases as the NPSH margin ratio is reduced towards 1.0 and below. This additional entrained air comes from the dissolved air coming out of solution as local static pressure drops below the vapor pressure. Centrifugal pumps not specifically designed to transport gas-liquid mixtures can generally accommodate (at inlet pressures near one atmosphere) up to approximately 2-percent gas volume in the inlet nozzle without appreciable effect.[61]

Larger quantities of entrained air can impact pump mechanical performance, including complete loss of prime or air binding and mechanical damage. Operation in an air-bound condition can cause overheating and failure (e.g., seizing of the impeller in the casing of the pump). This damage can occur in 10 minutes or less. The entrained air may be from air entrained in the suction water source, transported by vortices, or by dissolved air coming out of solution. The sump and suppression pool configurations should eliminate consideration of entrained air (e.g., because of air entrained by containment sprays) and vortices since any air bubbles will rise to the free surface of the pool and steps are taken in sump design to eliminate vortices. In addition, data developed as part of the resolution of USI A-43 show that vortices decay to negligible levels within 14 pipe diameters so that vortices created in a pool would not travel completely through the pump intake piping to the pump suction.[24]

Another concern with operating a pump in the vicinity of the 3-percent NPSHr condition is the damage that the water vapor or entrained air, or both, could do within the pump to the mechanical shaft seal faces, which could fail in a very short period of time if the seal faces run dry. Excessive entrained air tends to accumulate in the vicinity of the shaft, where the mechanical seal is housed. This means that, to protect the mechanical seal faces from this excess entrained air in the vicinity of the 3-percent NPSHr condition, dual mechanical seals with an external cold water flush system (or equal) should be provided.

6.3.6 Pump Flow Rate

The flow rate chosen for the NPSHa analysis should be greater than or equal to the flow rate assumed in the safety analyses that demonstrate adequate core and containment cooling. This ensures that the safety analysis and the NPSH analysis are consistent.

If the assumption that NPSHa equals $NPSHr_{eff}$ is used to determine the containment accident pressure employed, then the pump flow rate used in the core and containment cooling calculations should be equal to or less than the flow rate resulting from a 3-percent decrease in pump TDH.

6.3.7 Duration of the Need for Containment Accident Pressure

As stated above, based on pump performance considerations, the time for operation in the region of maximum cavitation erosion should be limited.

In addition, in considering containment integrity, the duration of the need for containment accident pressure to maintain acceptable NPSHa is, in general, not risk significant. Therefore, no time limit based on containment integrity is necessary since such factors as pre-existing leaks or failure to isolate the containment upon receipt of a containment isolation signal dominate risk and are independent of the time interval during which containment accident pressure is used.

6.3.8 Loss of Containment Isolation and Containment Leakage

The analysis should consider a loss of containment isolation that could compromise containment integrity. Possible losses of containment integrity include containment venting required by procedures or loss of containment isolation from a postulated Appendix R fire. It should be demonstrated conservatively that, for the plant examined, loss of containment integrity from these causes cannot occur or that they would occur only after use of containment accident pressure is no longer needed.

To reduce the likelihood of a preexisting leak, licensees proposing to use containment accident pressure in determining NPSH margin should do the following:

(1) Determine the minimum containment leakage rate sufficient to lose the containment accident pressure needed for adequate NPSH margin.

(2) Propose a method to determine whether the actual containment leakage rate exceeds the leakage rate determined in (1) above. For inerted containments, this method could consist of a periodic quantitative measurement of the nitrogen makeup performed at an appropriate frequency to ensure that no unusually large makeup of nitrogen occurs. Monitoring oxygen content is another method. For subatmospheric containments, a similar procedure might be used.

(3) Propose a limit on the time interval that the plant operates when the actual containment leakage rate exceeds the leakage rate determined in (1) above.

6.4 Overcooling the Containment during an Event in Which Containment Accident Pressure Is Used

The licensee should demonstrate that operation of sprays and fan coolers will not cause the containment accident pressure to be less than that needed to maintain adequate NPSHa. Operator action to control the containment pressure by means of containment sprays or fan coolers is acceptable, if justified. The appropriate procedures (e.g., emergency, abnormal) should include adequate guidance.

6.5 Quantifying NPSH Margin

One of the goals of this work is to quantify the margin between the expected (realistic, best-estimate, nominal) value of NPSH margin and the NPSH margin obtained from licensing calculations. This has been done in several ways.

For non-DBAs, termed “special events” for BWRs, realistic containment calculations are acceptable. This is consistent with NRC staff guidance for these events. Realistic calculations imply that no conservative bias is built into the calculations. Conservative assumptions such as the single-failure assumption are not necessary. Input values may be those associated with normal operation and not values based on technical specification limiting conditions for operation or bounding assumptions (e.g., 100-percent drywell relative humidity). If a realistic value is not available or cannot be easily defined, a more conservative value should be used. For example, the service water temperature may vary over a wide range (depending on the season); therefore, the service water temperature giving the more limiting NPSH margin should be used.

For the non-design-basis calculations, the NPSHr may be used without considering its uncertainty.

For DBAs, a Monte Carlo statistical NPSH margin analysis should be used to quantify NPSHa. This should be compared with a realistic calculation.

For the design-basis calculations, the NPSHr used should include its uncertainty.

6.6 Guidance Summary

Based on the discussion in Section 3, the NRC staff proposes the following guidance for the use of containment accident pressure in determining NPSHa of safety-related pumps.

6.6.1 $NPSHr_{eff}$

For DBAs, a value of $NPSHr_{eff}$ should be used in the analyses concerning the use of containment accident pressure. $NPSHr_{eff}$ includes the uncertainty in the value of $NPSHr_{3\%}$ based on vendor testing and installed operation. The effects of motor slip, suction piping configuration, and air content, and wear ring leakage should be included.

$$NPSHr_{eff} = (1 + \text{uncertainty})NPSHr_{3\%}$$

For non-DBAs, $NPSHr_{3\%}$ may be used.

6.6.2 Maximum Pump Flow Rate for the NPSHa Analysis

The maximum flow rate chosen for the NPSHa analysis should be greater than or equal to the flow rate assumed in the safety analyses that demonstrate adequate core and containment cooling. This ensures that the safety analysis and the NPSH analysis are consistent. If the NPSHa is assumed to equal the $NPSHr_{3\%}$ (the usual assumption for determining the amount of containment accident pressure used), then the flow rate used in the core and containment cooling analyses should also be equal to or greater than the flow rate resulting from a 3-percent decrease in pump TDH.

6.6.3 Conservative Containment Accident Pressure for Calculating NPSHa

A 95/95 lower tolerance limit should be used to calculate the containment accident pressure used to determine the NPSHa.

6.6.4 Assurance that Containment Integrity is not Compromised

It should be demonstrated conservatively that, for the plant examined, loss of containment integrity from containment venting, circuit issues associated with an Appendix R fire, or other causes cannot occur or that they would occur only after use of containment accident pressure is no longer needed.

6.6.5 Operator Actions

Operator action to control containment accident pressure is acceptable. The NRC staff should approve any operator actions, and the appropriate plant procedures (e.g., emergency, abnormal) should include them.

6.6.6 NPSHa less than NPSHr or $NPSHr_{eff}$

It is possible that the NPSHa may be less than $NPSHr_{eff}$ (LOCA) or $NPSHr_{3\%}$ (non-DBA). Operation in this mode is acceptable if appropriate tests are done to demonstrate that the pump will continue to perform its safety functions. The following conditions should apply:

- Predicted operation during the postulated accident below $NPSHr_{eff}$ (LOCA) or $NPSHr_{3\%}$ (nondesign-basis event) is of limited duration (less than 100 hours).

- The tests are conducted on the actual pump with the same mechanical shaft seal (including flush system) or at least a pump of the same model, size, impeller diameter, materials of construction, and pump seal and flush system.

- The test is conducted at the same (field application) speed.

- The test is conducted at the actual predicted NPSHa since testing at a lower NPSHa can actually reduce, rather than increase, the cavitation erosion rate in some cases.

- The test duration should be for the time NPSHa is predicted to be less than $NPSHr_{eff}$ (LOCA) or $NPSHr_{3\%}$ (nondesign-basis event).

- The flow rate and discharge head must remain above the values necessary to provide adequate core and containment cooling.

6.6.7 Assurance of no Pre-existing leak

Licensees and applicants should consider a loss of containment isolation that could compromise containment integrity. Possible losses of containment integrity include containment venting required by procedures or loss of containment isolation from a postulated Appendix R fire. It should be demonstrated conservatively that, for the plant examined, loss of containment integrity from these causes cannot occur or that they would occur only after use of containment accident pressure is no longer needed.

To reduce the likelihood of a preexisting leak, licensees proposing to use containment accident pressure in determining NPSH margin should do the following:

(1) Determine the minimum containment leakage rate sufficient to lose the containment accident pressure needed for adequate NPSH margin.

(2) Propose a method to determine whether the actual containment leakage rate exceeds the leakage rate determined in (1) above. For inerted containments, this method could consist of a periodic quantitative measurement of the nitrogen makeup performed at an appropriate frequency to ensure that no unusually large makeup of nitrogen occurs. Monitoring oxygen content is another method. For subatmospheric containments, a similar procedure might be used.

(3) Propose a limit on the time interval that the plant operates when the actual containment leakage rate exceeds the leakage rate determined in (1) above.

6.6.8 Maximum Erosion Zone

The zone of maximum erosion rate should be considered to lie between NPSH margin ratios of 1.2 to 1.6. The permissible time in this range, for very-high-suction energy pumps, should be limited unless operating experience, testing, or analysis justifies a longer time. Realistic calculations should be used to determine the time within this band of NPSH ratio values.

6.6.9 Estimate of NPSH Margin

A realistic calculation of NPSHa should be performed to compare with the NPSHa determined from the Monte Carlo 95/95 calculation.

6.6.10 Assurance of Pump Operability for Total Time Required

The necessary mission time for a pump using containment accident pressure should include not only the duration of the accident when the NPSH margin may be limited, but any additional time needed for operation of the pump after recovery from the accident when the pump is needed to maintain the reactor or containment, or both, in a stable, cool condition but at a much greater NPSH margin. This additional time is usually taken as 30 days.

7.0 CONCLUSIONS

Use of containment accident pressure potentially degrades defense-in-depth by making a connection between two fission product barriers (the containment and the fuel) which ideally should be independent. The NRC staff has considered this acceptable based on (1) the stringent measures taken to ensure containment integrity, (2) robust pump designs, (3) conservative calculations for the design basis accident (LOCA), and (4) the low risk calculated by the staff and licensees.

Containment integrity is an extremely important part of reactor safety. Design, regulations, technical specifications, surveillances and plant procedures all combine to make the containment highly reliable. Because of the measures taken to ensure containment integrity, containment integrity is assumed in design-basis accident analysis and in several regulations. The regulations include 10 CFR 50 Appendix K and the regulations governing offsite dose calculations (10 CFR 50.67 and 10 CFR Part 100).

As recommended by ACRS, the staff has developed draft guidance to attempt to quantify both margin and uncertainty in NPSHa and NPSHr.

8.0 ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BEP	best efficiency point
BWR	boiling-water reactor
CCF	common cause failure
CDF	core damage frequency
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
DBA	design basis accident
ECCS	emergency core cooling system
EPRI	Electric Power Research Institute
GEH	General Electric Hitachi
GL	generic letter
Gpm	gallons per minute
GSI	generic safety issue
HI	Hydraulic Institute
HPCI	high pressure coolant injection
ILRT	integrated leakage rate test
L_a	containment leakage rate defined in 10 CFR 50 Appendix J

LERF	large early release fraction
LLRT	local leakage rate test
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
MSIV	main steam isolation valve
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NPSHa	available net positive suction head
NPSHr	required net positive suction head
$NPSHr_{3\%}$	required net positive suction head corresponding to a 3% reduction in the pump total dynamic head
$NPSHr_{eff}$	Effective required net positive suction head $= NPSHr_{3\%}$ (1 + uncertainty)
NPSH margin	NPSHa - NPSHr
NPSH margin Ratio	NPSHa/NPHr
NRC	U.S. Nuclear Regulatory Commission
N_S	specific speed
N_{SS}	suction specific speed
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
psig	pounds per square inch gauge
PWR	pressurized water reactor
RCIC	reactor core isolation cooling
RHR	residual heat removal
RWST	refueling water storage tank

SSC	systems, structures and components
SRP	Standard Review Plan
TDH	total dynamic head
USI	unresolved safety issue

9.0 REFERENCES AND NOTES

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ENCLOSURE 2

Areas of Agreement and Disagreement between the NRC Staff and the Advisory Committee on Reactor Safeguards (ACRS)

ADAMS ML102780592

Areas of Agreement and Disagreement between the U. S. Nuclear Regulatory Commission Staff and the Advisory Committee on Reactor Safeguards

The U.S. Nuclear Regulatory Commission (NRC) staff (staff) has reviewed Advisory Committee on Reactor Safeguards (ACRS) correspondence to the staff and the Commission. The areas where the staff agrees or disagrees with ACRS regarding containment accident pressure are discussed in the excerpts from that correspondence, below.

1. May 19, 2010, letter from ACRS to the NRC Executive Director for Operations (EDO) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101300332)

   ACRS: The draft guidance developed by the staff "provides an improved framework for a more comprehensive assessment of the acceptability of crediting containment accident pressure in meeting net positive suction head [NPSH] requirements." However, ACRS notes that the guidance is deterministic and "should be complemented by a plant-specific Probabilistic Risk Assessment (PRA) analysis of the impact of containment accident pressure (CAP) credit."

STAFF POSITION: The staff developed more detailed and quantitative guidance to respond to ACRS recommendations that margins and uncertainties be explicitly addressed and discussed it at the 572nd ACRS meeting on May 6, 2010. The staff intends to apply this guidance, after further discussions with the industry, to future reviews involving the use of CAP in determining NPSH margin.

Also, at the staff's request, the Boiling-Water Reactor Owners Group (BWROG) developed a statistical approach to calculating NPSH margin (described in Enclosure 1, Section 6.3.4) that also quantifies margin (ADAMS Accession No. ML080520261). The staff's recently-developed guidance contains portions of the BWROG approach.

The staff's judgment is that CAP credit does not call into question adequate protection of public health and safety. This is based on a licensee's or applicant's compliance with NRC regulations, the high standards for maintaining containment integrity and the low risk of crediting CAP as determined in probabilistic risk assessments. As discussed in the SECY paper, current Commission guidance is that the staff cannot require risk information in this case.

2. March 18, 2009, letter from ACRS to EDO (ADAMS Accession No. ML090700460)

   ACRS: For cases in which operator actions are required to maintain containment overpressure, licensees should show how these actions can be implemented in their procedures, that they can be performed reliably, and that any increase in risk associated with these actions is acceptably small.

STAFF POSITION: The staff agrees with this recommendation, except with respect to risk. Staff human factors experts review licensee or applicant proposals to use operator actions to perform safety functions to ensure that the actions can be safely performed and are consistent with regulations, staff guidance, and Commission policy.

3. March 18, 2009, letter from ACRS to EDO (ADAMS Accession No. ML090700460)

ACRS: Credit for CAP should be limited in amount and duration.

STAFF POSITION: The staff disagrees with this recommendation for several reasons. The important parameter for successful pump operation with respect to avoiding excessive pump cavitation is the NPSH margin. As discussed in Enclosure 1, Section 2.0, containment pressure is one consideration in determining NPSH margin. However, any limit on the amount of CAP is arbitrary since it does not, by itself, determine whether a pump will or will not cavitate excessively. The staff guidance is formulated in terms of successful operation of the emergency core cooling system (ECCS) and containment heat removal pumps, rather than indirectly by using the CAP. The staff guidance ensures that these pumps can perform their safety function with respect to cavitation by specifying limits on the NPSH margin, including uncertainties in pump parameters and containment accident analysis results. A limit on the time that a pump depends on CAP for acceptable performance is also arbitrary and not related to any physical failure mechanism.

4. March 18, 2009, letter from ACRS to EDO (ADAMS Accession No. ML090700460)

ACRS: Regulatory Guide 1.82 Revision 3 ["Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident," issued November 2003 (ADAMS Accession No. ML033140347)] should be revised to include thermal-hydraulic analyses, which address the conservatisms associated with the licensing-basis analyses and explicitly account for uncertainties and probabilistic risk assessment (PRA) results consistent in scope and quality with that specified by Regulatory Guide 1.174 ["An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued July 1998 (ADAMS Accession No. ML003740133)].

STAFF POSITION: The staff agrees with the explicit accounting for uncertainties, addressed above as Item 1. The guidance developed by the staff explicitly accounts for uncertainties in NPSH margin.

Regarding the use of PRA, the staff's judgment is that CAP credit does not call into question adequate protection of public health and safety. As discussed in the SECY paper, current Commission direction is that the staff cannot require risk information in this case.

5. March 18, 2009, letter from ACRS to EDO (ADAMS Accession No. ML090700460)

ACRS: Licensees should continue to be requested to use the current guidance in Regulatory Guide 1.82, Revision 3, and the licensing-basis analyses assumptions and methods to demonstrate that the available net positive suction head (NPSH) exceeds that required for operation of the emergency core cooling system (ECCS) and containment heat removal system pumps.

STAFF POSITION: The staff developed new guidance as ACRS recommended. When finalized, this new guidance will replace the guidance in Regulatory Guide 1.82, Revision 3. The staff is currently discussing the new guidance with BWROG. The new methods focus on pump performance and include some pump cavitation phenomena not previously considered by the industry or the staff in these types of reviews. Enclosure 1, Section 6.0, describes this guidance.

6. ACRS letter dated September 20, 2005 (ADAMS Accession No. ML052630562)

ACRS: Credit for containment accident pressure should only be granted for robust containments for which there is a positive means for indication of containment integrity such as inerted or subatmospheric containments.

STAFF POSITION: The staff disagrees with this recommendation. A leakage rate greater than the value specified in the plant's technical specifications is beyond a plant's design basis. However, the importance of this recommendation will be evaluated as part of the suggested backfit analysis if the Commission directs the staff to pursue Option 2.