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USNRC

September 3, 2004 (10:30AM)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE

August 30, 2004

Office of the Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rulemaking and Adjudications Staff

Re: Docket No. 50-271 - Extended Power Uprate at Vermont Yankee Nuclear Power Station

Dear Sir/Madam:

Please find enclosed for filing an original and two copies of the Vermont Department of Public Service Notice of Intention to Participate and Petition to Intervene with Exhibits, Affidavit of William K. Sherman, Notice of Appearance from Sarah Hofmann and Anthony Z. Roisman, and Certificates of Service.

Service may be made on the Vermont Department of Public Service at the following:

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If you have any questions about this filing, please call me at 802-828-3088. Thank you for your assistance in making this filing.

Very truly yours,

Sarah Hofmann

Sarah Hofmann
Special Counsel

Template = SECY-037

SECY-02

cc: Lawrence Chandler, Esq.
John Fulton, Esq.
Jay Silberg, Esq.
Anthony Z. Roisman, Esq.

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In Re: Entergy Nuclear Vermont Yankee LLC and Entergy Nuclear Operations, Inc.	**)))**	**Docket No. 50-271 (Extended Power Uprate)**

NOTICE OF APPEARANCE

Pursuant to 10 CFR §2.314(b) Sarah Hofmann and Anthony Z. Roisman file this Notice of Appearance on behalf of the Vermont Department of Public Service, which is the single designated representative for the State of Vermont for the above-entitled proceeding:

Sarah Hofmann, Esq.
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Ms. Hofmann is an employee of the State of Vermont as a full-time Special Counsel to the Department of Public Service. She is a member of the Vermont Bar. Mr. Roisman is in private practice and has been retained by the Department of Public Service to assist in this matter. He is a member of the Bars of New York, the District of Columbia and Vermont.

Respectfully submitted,

Sarah Hofmann
Special Counsel
Department of Public Service
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Anthony Z. Roisman
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84 East Thetford Rd.
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August 30, 2004

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	Docket No. 50-271
Entergy Nuclear Vermont Yankee LLC	)	(Extended Power Uprate)
and Entergy Nuclear Operations, Inc.	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Notice of Appearance have been served upon the following persons by U.S. Mail, first class, or electronic mail as indicated.

VIA U.S. Mail:
Lawrence J. Chandler, Esq.
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Mail Stop - O-15 D21
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Washington, DC 20555-0001

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Sarah Hofmann, Special Counsel
Vermont Department of Public Service

Dated this 30th day of August, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

September 3, 2004 (10:30AM)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

In Re: Entergy Nuclear Vermont Yankee LLC and Entergy Nuclear Operations, Inc.	)))	Docket No. 50-271 (Extended Power Uprate)

**VERMONT DEPARTMENT OF PUBLIC SERVICE
NOTICE OF INTENTION TO PARTICIPATE
AND PETITION TO INTERVENE**

Filed on August 30, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In Re: Entergy Nuclear Vermont Yankee	**)**	
LLC and Entergy Nuclear	**)**	**Docket No. 50-271**
Operations, Inc.	**)**	**(Extended Power Uprate)**

VERMONT DEPARTMENT OF PUBLIC SERVICE NOTICE OF INTENTION TO PARTICIPATE AND PETITION TO INTERVENE

INTRODUCTION

The State of Vermont has consistently pursued issues related to safety at the NRC while reviewing other issues such as economic interests at the state level. The Vermont Department of Public Service ("DPS") has sent two letters to the NRC[1] requesting answers to the State's questions regarding the change in licensing basis to allow the crediting of containment overpressure for calculating certain pump net positive suction head ("NPSH") following postulated loss of coolant accidents ("LOCA"), station blackouts, and Appendix R fire events. Additionally, we are pleased that the issues associated with power uprate are being explored in the engineering inspection presently being undertaken at Vermont Yankee Nuclear Power Station ("Vermont Yankee") pursuant to Temporary Instruction 2515/158. The State supports this inspection, and believes the findings from this assessment may create the need to file new or amended contentions. However, despite the correspondence and the ongoing assessment, the State has not received answers that satisfy the State's concerns regarding the issue of taking credit for containment overpressure. Accordingly, the State of Vermont, to ensure the continued

[1]See letters attached from the Department of Public Service dated December 8, 2003 and June 8, 2004 (DPS Exhibits 13, 19).

safety of its citizens, must request a hearing to resolve its concerns and all of the contentions set forth below.

Vermont Yankee is located within the boundaries of the State of Vermont. DPS is the single representative of the State of Vermont for this Hearing. Therefore, pursuant to 10 CFR §2.309(d)(2), DPS is deemed to have standing for purposes of this proceeding and no further showing is required by DPS on that issue.

I. PARTICIPATION AS A MATTER OF RIGHT

The Atomic Energy Act, 42 U.S.C. §2021(l) specifies that "[w]ith respect to each application for Commission license authorizing an activity as to which the Commission's authority is continued pursuant to subsection (c) of this section", which subsection includes a license authorizing, *inter alia*, "the construction and operation of any production or utilization facility"[2] the NRC "shall afford reasonable opportunity for State representatives to offer evidence, interrogate witnesses, and advise the Commission as to the application". 42 U.S.C.

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[2] There cannot be any serious question that the application now pending to increase the thermal power of Vermont Yankee by 20% is a request to authorize operation of the plant at that level and falls within the scope of 42 U.S.C. §2021(c)(1) and (l). There is no need at this time to address the question of whether this language applies equally to all operating license amendments regardless of whether they seek to alter the power level or term of the operating license. In addition, the provisions of 10 CFR §50.91, which impose certain restrictions on state participation, are inapplicable here. That Section is limited to a Notice of Proposed Action under 10 CFR §2.105 which is deemed by the Commission to present no significant hazards. This is a Notice of Hearing for Consideration of Issuance of Amendment under 10 CFR §2.104.

§2021(c)(1) and (l).[3] 10 CFR §2.315(c) acknowledges these rights of a state in those cases where a hearing is being held. However, the statute extends the right to offer evidence and interrogate witnesses to all applications, even if pursuant to 10 CFR §2.309 no hearing will otherwise be held. Thus, in the case of a State and/or its designated representative, NRC must provide these rights of participation regardless of the existence of any "admissible contention" and include the right to present evidence and interrogate witnesses as to matters relevant to the application. DPS recognizes that without pre-filed contentions, witnesses may have difficulty preparing to answer questions posed and the Applicant, and Staff, if it participates, may have difficulty focusing their attention on the issues of concern to the State. For that reason DPS is submitting a statement of the contentions it now believes should be examined at the hearing and will supplement that list of contentions when and if new evidence, such as the report of the Engineering Inspection now being conducted at Vermont Yankee at the request of Vermont Governor James Douglas and the Vermont Public Service Board, becomes available.

DPS believes the most efficient manner by which these statutory rights can be exercised is to allow both depositions and live testimony to the extent the issues are not fully developed in the deposition, but should the NRC conclude all state interrogation must be conducted at a Board supervised hearing, DPS will conduct all of its interrogation of witnesses at that time. Although not specifically mentioned in §2021(l), DPS also believes that cross-examination of witnesses by

[3] Thus, DPS should not be required in this case to separately demonstrate that the provisions of Subpart G should apply to any Contentions which are admitted. Nonetheless, out of an abundance of caution, DPS provides that demonstration in the following paragraphs.

it will be more efficient if DPS submits cross-examination outlines, five days before the examination, to alert each witness to the subjects which DPS will explore. Similarly, DPS should have the right to seek production of documents if for no other reason than that production of documents will facilitate interrogation of witnesses and narrow the scope of their examination. Otherwise, witnesses will be asked questions about issues which are addressed in documents which either are not present during the interrogation or the analysis of which will require a hiatus in the interrogation.

DPS realizes that it may have information which Applicant, Staff or any other parties which may be permitted hearing status will want to see and although not required to do so by statute, will respond to reasonable requests for production of documents and is willing to have its witnesses cross-examined by Applicant, Staff or any admitted party provided outlines of cross-examination are submitted at least five days in advance for the witness to be prepared to fully answer the questions posed.

The following discussion follows the provisions of 10 CFR §§2.309 and 2.310 for purposes of simplicity and to demonstrate that even if DPS were not entitled to an adjudicatory hearing as a matter of right as to all of its contentions, it would nonetheless be entitled to an adjudicatory hearing on all these contentions under the provisions relevant to other parties.

II. PETITION TO INTERVENE

Pursuant to 10 C.F.R. §2.309 and the Notice of Consideration of Issuance of Amendment to Facility Operating License for Extended Power Uprate and Opportunity for a Hearing (TAC No. MC0761)(Notice) Petitioner, the DPS hereby submits contentions regarding Vermont Yankee's application for a license amendment to increase the approved thermal power at its boiling water nuclear power plant in Vernon, Vermont by 20% (uprate). As demonstrated below, these contentions should be admitted because they satisfy the NRC's admissibility requirements in 10 C.F.R. § 2.309.[4] Also, the State requests, and is entitled to, as demonstrated below, a full adjudicatory hearing with all the rights of discovery and cross-examination provided by 10 CFR Subpart G because DPS has met the requirements of 10 CFR 2.310 (d).[5]

A. CONTENTIONS, BASES AND SUPPORTING EVIDENCE

DPS submits the following contentions, bases and supporting evidence regarding the proposed Vermont Yankee uprate:

---

[4] Although these contentions meet the requirements of 10 CFR §2.309, DPS does not concede the procedures are lawful and reserves the right to challenge, in an appropriate legal forum, these procedures, as applied to DPS in this case, should that be necessary to permit DPS to present and fully adjudicate the important nuclear safety issues raised in its contentions.

[5] Although DPS meets the requirements of 10 CFR §2.310(d) for a full adjudicatory hearing on all contentions it raises, DPS does not concede the procedures of 10 CFR §2.310 which restrict use of full adjudicatory hearing procedures are lawful and reserves the right to challenge, in an appropriate legal forum, these procedures, as applied to DPS in this case, should that be necessary to permit DPS to fully adjudicate the important nuclear safety issues it raises.

First Contention

Applicant Has Claimed Credit for Containment Overpressure in Demonstrating the Adequacy of ECCS Pumps for Plant Events Including a Loss of Coolant Accident in Violation of 10 C.F.R. §50, Appendix A, Criteria 35 and 38[6] and Therefore Applicant Has Failed to Demonstrate That the Proposed Uprate Will Not Create a Significant Hazard as Required by 10 C.F.R. §50.92 and Will Not Provide Adequate Protection for the Public Health and Safety as Required by 10 C.F.R. §50.57(a)(3).

Bases

1. The portion of NRC Regulatory Guide 1.82, Revision 3 (DPS Exhibit 2) which purports to authorize containment overpressure credit has never been properly evaluated or approved by the Advisory Committee on Reactor Safeguards ("ACRS") in violation of the requirements of 42 U.S.C. §2039.

2. Regulatory Guide 1.82, Revision 3 is substantively indefensible because its authorization for the use of containment overpressure to demonstrate the NPSH required to properly operate ECCS pumps, improperly eliminates NRC safety requirements for defense in depth by multiple fission product barriers by allowing one barrier failure - containment failure - to compromise the effectiveness of two critical safety systems - containment and ECCS pump operation and eventually compromise the two remaining fission product barriers, fuel cladding and the reactor coolant system..

3. Even if Regulatory Guide 1.82, Revision 3, were applicable to this case, Applicant has

---

[6] Vermont Yankee is committed to the draft general design criteria published July 11, 1967 (32 FR 10213) (DPS Exhibit 1). The corresponding criteria are Draft Criteria 44 and 52.

failed to demonstrate that it meets the very limited condition required by the Regulatory Guide for use of containment overpressure in calculating NPSH for ECCS pump operation. In particular, Applicant has not shown and cannot show that use of containment overpressure in calculating NPSH for ECCS pump operation is either "necessary" or that plant operations or equipment cannot be "practicably altered" either by limiting thermal output of the reactor or upgrading the ECCS pumps.

Supporting Evidence

1. In issuing Regulatory Guide 1.82, Revision 3, the NRC has accomplished a major policy change regarding containment overpressure credit. NRC policy was previously clear in Safety Guide (Regulatory Guide) 1.1 (DPS Exhibit 3) that credit for containment overpressure was not allowed. Regulatory Guide 1.82 establishes a new criteria:

> **2.1.1.1** ECC and containment heat removal systems should be designed so that adequate available NPSH is provided to the system pumps, assuming the maximum expected temperature of the pumped fluid and no increase in containment pressure from that present prior to the postulated LOCAs. (See Regulatory Position 2.1.1.2.)
>
> **2.1.1.2** For certain operating BWRs for which the design cannot be practicably altered, conformance with Regulatory Position 2.1.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. Calculation of available containment pressure should underestimate the expected containment pressure when determining available NPSH for this situation. Calculation of suppression pool water temperature should overestimate the expected temperature when determining available NPSH.

This new criteria retains the restriction for crediting containment pressure, but alleviates

this restriction under certain conditions. Alleviation is not granted unless the "design cannot be practicably altered."

2. This major policy change has not received adequate review by NRC. Rather, the policy change is embedded in a detailed technical regulatory guide which is primarily focused on a different safety issue. Regulatory Guide 1.82, Rev. 3, *Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident,* was first issued as Regulatory Guide 1.82 (Rev. 0) in June 1974 with the title, *Sumps for Emergency Core Cooling and Containment Spray Systems* (DPS Exhibit 4). It is known throughout the industry as NRC's policy document addressing continuing unresolved safety issues regarding containment sump design, pump suction strainer design and debris loading assumptions. Background for these unresolved safety issues may be found in:

> Documents related to Unresolved Safety Issue (USI) A-43, *Containment Emergency Sump Performance* (DPS Exhibit 5)
>
> NRC Bulletin 96-03, *Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors* (DPS Exhibit 6)
>
> Documents related to Generic Safety Issue (GSI) 191, *Assessment of Debris Accumulation on PWR Sump Pump Performance* (DPS Exhibit 7)
>
> NRC Bulletin 2003-01, *Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors* (DPS Exhibit 8)

Therefore, Regulatory Guide 1.82 is known as a technical document for containment sumps. It is not a document in which a major change in policy to allow analytical crediting for containment pressure would be expected to reside.

3. This major policy change has not received the required review by the ACRS. The Atomic Energy Act requires ACRS to review and advise the NRC on proposed reactor safety standards:

> There is established an Advisory Committee on Reactor Safeguards consisting of a maximum of fifteen members appointed by the Commission for terms of four years each. The Committee shall review safety studies and facility license applications referred to it and shall make reports thereon, shall advise the Commission with regard to the hazards of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards.

42 U.S.C. §2039.

4. While the both the full ACRS and the ACRS Thermal-Hydraulic Phenomena Subcommittee reviewed the draft of Regulatory Guide 1.82, Rev. 3, before its issue, their review concentrated only on the technical issues of containment sump design, pump suction strainer design and debris loading assumptions, which have been so prominent throughout the last 30 years. Their review did not consider the containment overpressure policy change. NRC staff presented the policy change to both subcommittee and full committee almost as an afterthought.

> Another thing is Reg. Guide 1.1 has been subsumed into this current version. So only for some older plants they have to refer back to this Reg. Guide 1.1. For future plants, they refer to Reg. Guide 1.82 now for the NPSH issue.
>
> Next slide, please.

NRC staff presenter, T.Y Chang, ACRS Thermal-Hydraulic Phenomena Subcommittee transcript of August 20, 2003, at 21-22 (DPS Exhibit 9).

> Finally, within this version of the Reg Guide, another Reg Guide is

> subsumed into this one. That is Reg Guide 1.1, the net positive suction head for ECCS and containment heat removal system pumps. So Reg Guide 1.1 will no longer be in existence. It will be part of Appendix A of this Reg Guide.

NRC staff presenter, T.Y Chang, ACRS Full Committee, transcript of September 11, 2003, at 354 (DPS Exhibit 10). Since Dr. Chang did not note in the presentation that a critical portion of Reg. Guide 1.1 had been altered, it is not surprising that no ACRS member asked questions of Dr. Chang about containment overpressure credit following his presentation. The subject of the major containment overpressure credit policy change was not brought up again by any NRC presenter, nor did any ACRS member question the change through the lengthy investigation of the proposed Regulatory Guide. The investigation focused only on the technical details of containment sump design, pump suction strainer design and debris loading assumptions. See Subcommittee transcript of August 20, 2003, at 4-198 (DPS Exhibit 9), and Full Committee transcript of September 11, 2003, at 344-415 (DPS Exhibit 10).

5. The ACRS letter of September 30, 2003 (DPS Exhibit 11), that recommends issuing Regulatory Guide 1.82, Rev. 3, is similarly silent regarding the major policy change regarding containment overpressure credit. This supports a conclusion that the ACRS was not fully aware of the major policy change or its implications. This recommendation letter is long, filled with technical details and reservations about containment sump design, pump suction strainer design and debris loading assumptions. One may also conclude that the ACRS recommends issuing Regulatory Guide 1.82, Rev. 3, begrudgingly "in order to facilitate licensee response and resolution of technical issues." Letter at 1. The following is NRC staff M. Mayfield's request

for ACRS to recommend issuing Regulatory Guide 1.82, Rev. 3:

> NEI is preparing guidance that's more detailed than what you'll find in this regulatory guide. The staff will review that guidance, and we have yet to -- we and NRR will review that guidance document once NEI has it. And the decision will be made at that time, what vehicle to use to endorse that guidance, assuming that that's the direction we go. But in the interim, we felt like it was important to finalize this guide and get it on the street.

Full Committee transcript of September 11, 2003, at 346. By this, it is shown that Regulatory Guide 1.82, Rev. 3, is considered more as interim technical guidance, necessary to be "on the street," rather than a major policy change to allow containment overpressure credit.

6. Granting containment overpressure credit, as requested by the Applicant for Vermont Yankee power uprate, is an inappropriate encroachment on the historical defense-in-depth philosophy of the NRC, and similarly an encroachment on the appropriate application of defense-in-depth in the risk-informed regulatory environment. The history of defense-in-depth consideration was summarized by ACRS:

> Defense in depth is a nuclear industry safety strategy that began to develop in the 1950s. A review of the history of the term indicates that there is no official or preferred definition. Where the term is used, if a definition is needed, one is created consistent with the intended use of the term. Such definitions are often made by example.
>
> In a 1967 statement submitted to the Joint Committee on Atomic Energy by Clifford Beck, then Deputy Director of Regulation for the Atomic Energy Commission, three basic lines of defense for nuclear power reactor facilities were described. The first line was the prevention of accident initiators through superior quality of design, construction and operation. The second line was engineered safety systems designed to prevent mishaps from escalating into major accidents. The third line was consequence-limiting safety systems designed to confine or minimize the escape of fission products to the environment.

> A 1969 paper by an internal study group of the Atomic Energy Commission identified the issue of balance among accident prevention, protection, and mitigation, with the conclusion that the greatest emphasis should be put on prevention, the first line of defense.
>
> A 1994 NRC document identifies the elements of the defense in depth safety strategy as accident prevention, safety systems, containment, accident management, and siting and emergency plans. Other interpretations of defense in depth can be found in INSAG-3 and INSAG-10
>
> The historical record indicates an evolution of the term from a narrow application to the multiple barrier concept to an expansive application as an overall safety strategy. The term has increased in scope and gained stature over time. The history also indicates that defense in depth is considered to be a concept, an approach, a principle or a philosophy, as opposed to being a regulatory requirement per se.
>
> Currently the term is commonly used in two different senses. The first is to denote the philosophy of high level lines of defense, such as prevent accident initiators from occurring, terminate accident sequences quickly, and mitigate accidents that are not successfully terminated. The second is to denote the multiple physical barrier approach, most often exemplified by the fuel cladding, primary system, and containment.
>
> One of the essential properties of defense in depth is the concept of successive barriers or levels. This concept applies equally well to multiple physical barriers and to high level lines of defense. A closely related attribute would be requiring a reasonable balance among prevention, protection and mitigation.

ACRS Paper, *On the Role of Defense in Depth in Risk-informed Regulation*, attached to ACRS letter, May 19, 1999, *The Role of Defense in Depth in a Risk-informed Regulatory System* (DPS Exhibit 12). Therefore, historically, the containment is one of the three multiple physical barriers. However, under the conservative assumptions of historical regulatory evaluation, if containment overpressure credit is granted for ECCS pump NPSH, and then the containment barrier fails, the

following is the result. The ECCS pump dependency on the containment means that, were the containment to fail, the ECCS pumps would also be assumed to fail, and this would result in failing the fuel cladding barrier and the primary system barrier if it was not already failed by the initiating event. Creating the dependency between containment functioning and ECCS pump functioning voids the historical multiple physical barrier defense-in-depth strategy.

7. Defense-in-depth by multiple physical fission product barriers is integral to and embedded in NRC regulations. See 10 C.F.R. §50, Appendix A, Criteria 10 through 19 which are labeled, *Protection through Multiple Fission Product Barriers.* This defense-in-depth concept recognizes that, while the licensing basis assumes a single failure, real accidents and events do not proceed according to planned scenarios and often involve multiple failures. Therefore, if the reactor coolant system barrier fails, despite stringent design control provisions, the fuel cladding and reactor containment barriers prevent fission product release. After the reactor coolant barrier is breached, either through LOCA or through the requirement to control pressure with relief valves, if the fuel cladding fails despite ECCS systems which are designed to prevent such failure, then the reactor containment prevents fission product release. Conversely, if the reactor containment fails despite design provisions to prevent such failure, the fuel cladding is provided to stay intact and prevent fission product release. The key to effective defense-in-depth through multiple fission product barriers is not to create dependencies such that the failure of one barrier will lead to the failure of other barriers.

8. The policy change to allow ECCS pumps to rely on containment pressure creates a

dependency such that, in that condition, containment failure would lead to ECCS pump failure, which in turn would defeat cooling to the reactor and lead to fuel cladding and reactor coolant system failure.

9. In the above referenced letter, the ACRS summarized the emerging regulatory consideration of defense-in-depth:

> The most recent NRC policy statement that deals with defense in depth is the Probabilistic Risk Assessment (PRA) Policy statement published in 1995, which states, in part:
>
> > "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."
>
> The policy statement, thus, places PRA in a subsidiary role to defense in depth.
>
> In 1998, the NRC published Regulatory Guide 1.174. This guide establishes an approach to risk-informed decision making, acceptable to the NRC staff, which includes the provision that proposed changes to the current licensing basis must be consistent with the defense in depth philosophy. The RG 1.174 discussion states that, "The defense in depth philosophy . . . has been and continues to be an effective way to account for uncertainties in equipment and human performance." The discussion goes on to say that PRA can be used to help determine the appropriate extent of defense in depth, which, by example, is equated to balance among core damage prevention, containment failure prevention and consequence mitigation. The regulatory guide thus addresses the concern of preventing risk-informed regulation from undermining defense in depth. Defense in depth is primary, with PRA available to measure how well it has been achieved.

ACRS Paper, *On the Role of Defense in Depth in Risk-informed Regulation*, attached to ACRS letter, May 19, 1999, *The Role of Defense in Depth in a Risk-informed Regulatory System* (DPS

Exhibit 12). ACRS makes it clear in their summary that "[d]efense in depth is primary," and "PRA [is] in a subsidiary role to defense in depth." Therefore, voiding the multiple barrier philosophy by creating a dependency between the containment and the other two barriers violates one of the most fundamental and long-standing nuclear safety principles.

10. ACRS further elaborates in their May 19, 1999, letter regarding defense-in-depth:

> Defense in depth can still provide needed safety assurance in areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain.

By this criteria, granting overpressure credit that creates a common failure mode among the three multiple fission product barriers violates safety principles on two counts:

The first area of modern analysis that is poorly treated and with results quite uncertain is the area of risk evaluation (e.g., the potential impact on core damage frequency). The Applicant's risk evaluation calculates there is hardly any increase in risk from taking credit for containment overpressure. There reason for this result is that the risk evaluation used by Applicant is not sufficiently developed to properly evaluate the risk impact associated with granting this overpressure credit. The Applicant's risk evaluation uses nominal or average values of temperatures, pressures, flows and other parameters, rather than conservative values. Under this nominal value evaluation, torus temperatures do not rise enough to require containment overpressure. Therefore, there is no calculated additional risk associated with overpressure. However, this result is counter intuitive and incorrect. There is some probability that temperatures, pressures, flows and other parameters will be at conservative values, and that, if

containment failed in this situation, it would cause ECCS pump failure and increased core damage frequency, and therefore increased risk. However, risk evaluation techniques only assume nominal values and are not equipped to assign probabilities for a range of operating values. Therefore, the analytical technique does not properly calculate the increased risk from containment overpressure credit.

Second, in recommending issuing Regulatory Guide 1.82, Rev. 3, the ACRS summarized the state of modern analysis for ECCS pump NPSH without considering the containment overpressure issue. (It is shown above that ACRS recommended issuing the Regulatory Guide primarily to get the information "on the street.") ACRS concludes:

> The technical basis for analyzing the phenomena described in RG 1.82 is not mature, the available information is inconsistent, and the knowledge base is evolving. Therefore, it is likely that the licensees' responses will be disparate and difficult to evaluate unless more consistent guidance is developed.
>
> The zone of influence (ZOI) models need revision and resolution of inconsistencies.
>
> Neither RG 1.82 nor the knowledge base report (Ref. 2) gives adequate consideration to chemical reactions.

ACRS letter, September 30, 2003, *Draft Final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-coolant Accident."* (DPS Exhibit 11) (See also information provided for Contention II.) These ACRS conclusions show that ACRS has questions about the analytic techniques that are not resolved by the Regulatory Guide and which remain open questions. These conclusions show that the issuance of the

Regulatory Guide does not resolve all analytical issues, and that the calculation of the NPSH for ECCS pumps should be considered "poorly treated by modern analyses."

11. Although it cannot be concluded that NRC, and specifically ACRS, adequately considered the major policy change of granting overpressure credit, the limits for granting this credit in Regulatory Guide 1.82, Rev. 3, are very narrow. In the discussion section of the Regulatory Guide, it is stated:

> Predicted performance of the emergency core cooling and the containment heat removal pumps *should be* independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions... However, for some operating reactors, credit for containment accident pressure *may be necessary*. This should be minimized to the extent possible.

Regulatory Guide 1.82, Rev. 3, at 8 (Emphasis added). It is further stated:

> For certain operating reactors for which the design cannot be *practicably altered*, compliance with Regulatory Position 2.1.1.1 [i.e., no credit for containment accident pressure] may not be possible.

Regulatory Guide 1.82, Rev. 3, at 20 (Emphasis added). As shown below, the Application for power uprate requesting overpressure credit contains no showing that such credit is *necessary* nor that the uprate level or plant design cannot be *practicably altered* to avoid taking overpressure credit.

12. Regarding the *necessary* test, there is no apparent compelling reason that requires the Applicant to request a 20% power uprate of Vermont Yankee. Vermont Yankee is performing adequately and economically at its current power level. There is no power shortage in New

England. There is no way that Vermont Yankee's power 20% uprate could be found to be necessary. The need for containment overpressure credit can be eliminated by reducing the level of power uprate to a level that would not require overpressure credit, even to the current licensed power level. In the DPS December 8, 2003, letter to the NRC Staff (DPS Exhibit 13) , we asked:

> At what uprated power level could Vermont Yankee operate and not claim credit for containment accident pressure in its NPSH calculations?

Letter at 3. NRC responded on June 29, 2004 (DPS Exhibit 14):

> [T]he NRC staff has not performed calculations to determine the power at which containment pressure is not required when using conservative assumptions and the licensee has not presented such analysis to us.

Response at 5. From this it is clear there has been no consideration of the *necessary* test and no attempt to demonstrate that the 20% uprate is *necessary*. The Staff has not attempted to investigate this possibility by sending a Request for Additional Information (RAI) to Applicant to identify the highest power level at which credit for containment overpressure is not required. Furthermore, it is clear from the following NRC response in the June 29, 2004, letter that it ignores the *necessary* test altogether:

> DPS Question 2.a.2
>
> Does the agency believe that it is *necessary* to operate at extended uprated power level, thereby creating the necessity for allowing credit for containment accident pressure? If the answer is in the affirmative, please identify the reason the agency thinks operating at extended uprated power level is *necessary*?
>
> NRC Response to DPS Question 2.a.2

> The NRC staff makes no judgment on whether a proposed license amendment, such as a power uprate request, is necessary . . .

Response at 4. Since Applicant has made no attempt to demonstrate that it meets the pre-conditions for use of containment overpressure, it has not demonstrated that it qualifies to use such overpressure under the limited circumstances authorized by Regulatory Guide1.82, Rev. 3.

13. Regarding the *practicably altered* test, Applicant has not investigated or attempted to apply this test, either. Vermont Yankee design does not need to be *practicably altered* because containment overpressure credit is not required at its current licensed power level and neither is power uprate required. However, given that Applicant wants to implement the 20% power uprate, it has not shown that it is not possible to modify existing ECCS pumps or provide new ECCS pumps that do not require credit for containment overpressure in order to function. Neither has the NRC sent RAI's to investigate this possibility. Vermont witness, William Sherman, testified before the Vermont Public Service Board that the cost of Applicant's proposed power uprate is approximately $20/MWh or 2.0 cents per kWh. Docket No. 6812, Prefiled Direct Testimony, May 9, 2003, at 11 (DPS Exhibit 15). Since market power costs are at approximately 5.0 cents per kWh, Applicant will earn millions of dollars annually from the 100 MW uprate, clearly sufficient to *practicably alter* the ECCS pumps to function without crediting containment overpressure. Applicant has not shown that its ECCS pumps cannot be *practicably altered* to avoid the extraordinary design basis change of crediting containment overpressure.

Second Contention

Because of the Current Level of Uncertainty Associated with the Demonstration of the Adequacy of ECCS Pumps, Applicant Has Not Demonstrated That Allowing a Radical Departure from the Defense in Depth Principle Which Prohibits Use of Containment Overpressure to Provide the Necessary NPSH for ECCS Pumps Will Not Constitute a Significant Hazard (10 C.F.R. §50.92) and Will Provide Adequate Protection for the Public Health and Safety as Required by 10 C.F.R. §50.57(a)(3).

Bases

1. There is no reliable evidence of the magnitude of the impact of strainer and debris losses on pressure at the ECCS pumps following a LOCA.

2. Without sufficient information to adequately bound the uncertainties associated with the extent to which pressure at the ECCS pumps will be reduced following a LOCA, there is no reliable basis to justify using the equally uncertain containment overpressure to compensate for the unquantifiable pressure losses at the ECCS pump.

3. Vermont Yankee's current design basis and licensing basis recognize that containment pressure increases above atmospheric pressure for various plant events, but do not take credit for this increase in pressure to demonstrate that ECCS pumps will function properly. Thus, this increased containment pressure above atmospheric pressure serves as an additional safety margin or defense-in-depth for the functioning of ECCS pumps. It is inappropriate to abandon this safety margin or defense-in-depth by allowing containment overpressure credit because the calculations and analyses for determining NPSH of the ECCS pumps are uncertain and imprecise.

Supporting Evidence

1. The ACRS, in reviewing the role of defense-in-depth in a risk informed environment, stated:

> Defense in depth can still provide needed safety assurance in areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain.

ACRS Letter, May 19, 1999, *The Role of Defense in Depth in a Risk-informed Regulatory System* (DPS Exhibit 12).

2. Vermont Yankee Calculation VYC-0808, Rev. 6 (DPS Exhibit 16), was provided as Exhibit 1 to Attachment 4 of Supplement 8 of Applicant's request for extended power uprate. VYC-0808, Rev. 6 calculates the strainer and debris losses for the NPSH calculation. However, the calculation is not conservative because it does not incorporate all the provisions of Regulatory Guide 1.82, Rev. 3 (DPS Exhibit 2).

3. Even if Regulatory Guide 1.82, Rev. 3 were followed, there would not be high confidence in the calculated results. ACRS Thermal-Hydraulic Phenomena Subcommittee Chairman Graham Wallis, during the ACRS review, stated:

> The concern that I have is that you'll put out the Reg Guide, which I think is the right thing to do, get things moving, put out this Reg Guide and say, thou shalt evaluate all of these things.
>
> My concern is there are so many things which there isn't much of a technical basis for. That these folks may come back with some half-baked analysis, which gets accepted. Because nobody knows. And then further research now in progress reveals that it shouldn't have been accepted.

ACRS Full Committee, transcript September 11, 2003, at 387-8 (DPS Exhibit 10).

4. In response to Chairman Wallis, NRC staff presenter, M. Mayfield, admits the flaws and shortcomings of the analytical techniques in Regulatory Guide 1.82, Rev. 3:

> Well, that's why -- that is one of the downsides of confirmatory research where I live. The other thing I had said was that we have had, and continue to have, some discussions with NRR about how much more do they need to be comfortable to assess what the licensees are going to bring in the door. The reason for pushing it forward at this time, to include that loosely worded caveat or flag, is frankly let's put everything on the table at this time to what level of information we have. And so we felt like the itch is real, and we needed to flag it in this to the level of detail we can support today, which is to say this is something that should be evaluated. We will continue to work with NRR, looking at how much more information they need to support an evaluation. But today, we felt like we needed to at least flag the issue in the guide . . . The level of detail that we put in this is admittedly sparse.

Id, at 388-9. A little later on, Chairman Wallis again criticized draft Regulatory Guide 1.82, Rev. 3 in the following exchange:

> MEMBER WALLIS: This three-region two-phase conical jet model, with numbers on it Figure 17, comes from -- doesn't come from the Sandia work. It doesn't come from the one you referenced. The only place that I could find it was in a later new Reg [sic - NUREG] that the agency prepared.
>
> Right, and my personal view is that it's a complete misapplication of the Sandia work. Maybe, if my colleagues give me permission, I might actually make a presentation to them on that. But I just wanted to warn you -- I don't know if you've looked at its origin and seen if you believe it or not.
>
> DR. LETELLIER (NRC Contractor from Los Alamos National Lab): That model has been discredited by the Barsebaeck event.
>
> MEMBER WALLIS: Right, it has been.
>
> DR. LETELLIER: In fact --

> MEMBER WALLIS: And by practice it's been. But it's in your documents that you've accepted it.
>
> DR. LETELLIER: Are you referring to the knowledge base? Please interpret --
>
> MEMBER WALLIS: But it's there, as being authoritative.
>
> DR. CHANG (NRC Staff): The knowledge base report is trying to document order information and pass --
>
> MEMBER WALLIS: But without the critical evaluation, you know, leaves it up to the utilities or NEI to select what's suitable for their purposes.
>
> DR. LETELLIER: Well, that's a fair criticism, that it is presented as authoritative. But it's also intended to be historical.

Id., at 392-3.

5. This uncertainty and imprecision in strainer and debris analytical modeling that was exhibited in ACRS questions, is echoed in the letter recommending the issuing of Regulatory Guide 1.82, Rev. 3. The ACRS concluded:

> The technical basis for analyzing the phenomena described in RG 1.82 is not mature, the available information is inconsistent, and the knowledge base is evolving. Therefore, it is likely that the licensees' responses will be disparate and difficult to evaluate unless more consistent guidance is developed.
>
> The zone of influence (ZOI) models need revision and resolution of inconsistencies.
>
> Neither RG 1.82 nor the knowledge base report (Ref. 2) gives adequate consideration to chemical reactions.

ACRS letter, September 30, 2003, *Draft Final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-coolant Accident."* (DPS

Exhibit 11) These conclusions by the ACRS demonstrate that, even with the issuance of the new Regulatory Guide, important uncertainties in analytical methods still exist. These are examples of open and unresolved questions about the analytical methods for calculating the strainer head loss and debris loading effect. This demonstrates that, because of lack of confidence in analytical results, the defense-in-depth and safety margin inherent in not taking overpressure credit must be retained to provide reasonable assurance of adequate protection of the public health and safety.

6. Another reason that containment overpressure credit should not be granted is that there is insufficient conservatism and margin in the values used for required NPSH or NPSHr in Applicant's demonstration of ECCS pump adequacy. The values used for NPSHr are determined in calculation VYC-0808, Rev. 6, which identifies areas of imprecision and uncertainty. Both the residual heat removal and core spray pumps were only NPSH-tested over a limited flow range. No head drop was specified on the original curves. VYC-0808, Rev. 6, Attachment 5, p. 6 of 19. According to the pump vendor, the tests of the residual heat removal pumps were not complete enough to determine the exact NPSH-characteristics of the pumps. Id. No vibration readings were taken in the NPSH tests for the residual heat removal pumps. Id., Attachment 5, p. 7 of 19. Only one of the four residual heat removal pumps was tested for NPSHr, and this value was assumed correct for the other three pumps. Id., at 9. The core spray pumps original witness tests for NPSHr do not bracket the expected flow range during accidents. Id., at 10. NPSHr for the core spray pumps was not determined from Vermont Yankee's pumps, but rather for pumps for another customer not even the same size as Vermont Yankee's. The NPSHr for Vermont

Yankee core spray pumps was estimated by the vendor from this other pump rather than measured from Vermont Yankee's core spray pumps. Id. For both residual heat removal and core spray pumps, curve fit regimes were used to acquire NPSHr values for specific flow rates used in the demonstrations of adequacy. Id, at 12-13. Their curve fit programs create an uncertainty in the precision of results. The vendor summarized the state of NPSH testing:

> The original pump NPSH requirements were not well defined. The result was only two (2) NPSH-Test points for each capacity were measured. From two (2) NPSH-test points it is not possible to establish the "knee." At each NPSH-test point (during witness tests) the pumps were operating only a few minutes and the capacity-range was limited.

Id., Attachment 5, p. 10 of 19. In the vendor prepared document (Attachment 5 to VYC-0808, Rev. 6), there is no indication of accounting for instrumentation inaccuracies in test instruments. Nor is margin provided to account for the extrapolation of data and assumptions used for actual test data that is lacking.

7. The Hydraulic Institute recommends that margin be applied above measured NPSHr. The NRC staff asked about this margin in RAI SPSB-C-25 (DPS Exhibit 17), and Applicant responded as follows:

> The required NPSH (NPSHR) information provided for the Vermont Yankee Nuclear Power Station (VYNPS) core spray (CS) and residual heat removal (RHR) pumps by the manufacturer specifically address time-phased operational requirements with low available NPSH (NPSHA). No specific margin is included or required in the NPSHA calculation. However, there is some margin between the overpressure required and the credited overpressure requested and more margin to the overpressure available.

Entergy Request for Extended Power Uprate, Supplement 8, Attachment 2, page 183. Applicant

states that no margin is provided for measured NPSHr values and also states no margin is required in available NPSH. However, the uncertainties from instrument inaccuracies, extrapolations and assumptions instead of hard test data, direct that margin should be provided. While Applicant notes in response that the remaining containment pressure above the credited overpressure remains as margin, it is more appropriate to reserve the entire containment overpressure to allow for analytical uncertainties rather than take credit for some or all of it to seek to resolve the separate safety issue of NPSH following a LOCA.

8. Uncertainty also exists in the value that the Applicant uses for containment leakage. Frequently the as-found condition of containment isolation valves from their leakage tests exceeds allowables such that containment leakage is underestimated.

9. Analytical uncertainties also exist in the containment pressure and torus temperature calculations, and these uncertainties are another reason that containment overpressure should be retained as a safety margin and defense-in-depth. In Section 4.2.6 of Safety Analysis Report for Constant Pressure Power Uprate ("PUSAR") (DPS Exhibit 18), Applicant has stated it requires containment overpressure credit for loss of coolant accidents (LOCAs), station blackouts (SBOs), Appendix R fire events and anticipated transients without scram (ATWS). PUSAR is deficient since it does not identify the amount of overpressure developed or credited for the SBO, Appendix R fire events and ATWS, although the NRC staff has received this information through data requests. The calculations to develop containment pressure and torus temperature responses for these events are complex. For this reason, the DPS letter of June 8, 2004 (DPS

Exhibit 19), requests that the NRC staff perform independent verifications of Applicant's calculations for LOCAs, SBOs, Appendix R, and ATWS events. NRC has not responded to the DPS June 8, 2004 letter. However, based on RAI's, it appears NRC is only independently verifying the LOCA calculations. If this is the case, this will leave uncertainty regarding the accuracy of the SBO, Appendix R and ATWS calculations.

10. Even if NRC's independent verification of LOCA calculations confirm the results of Applicant's calculations, uncertainty will still exist in the calculations. The scrutiny on LOCA calculations has resulted in two modifications from the results provided in PUSAR in a period of less than a month. On July 1, 2004, Applicant corrected VYC-0808, Rev. 6 with change notice 5 (DPS Exhibit 20) to incorporate the revised containment leak rate for power uprate. Entergy Request for Extended Power Uprate, Supplement 8, Attachment 4, Exhibit 1. On July 16, 2004, Applicant again corrected VYC-0808, Rev. 6 with change notice 6 (DPS Exhibit 21) to use a conservative containment spray thermal mixing efficiency. Entergy Request for Extended Power Uprate, Supplement 9, Attachment 2. It is likely that additional calculation changes will be discovered with further review and as time goes on. These results indicate that uncertainty exists within the analytical methods such that it is appropriate to retain the entire containment overpressure as a safety margin and defense-in-depth.

11. The above information shows that significant uncertainties exist in 1) the method of calculating strainer losses and debris loading effects, 2) the proper value of the required NPSH, 3) the value used for containment leakage, and 4) the results of calculations that have unverified

input parameters and calculation methods. These latter calculations have a recent history of revision by the Applicant when viewed carefully. All of these uncertainties lead to the conclusion that the ACRS statement in its paper on Defense in Depth must be accepted. Defense in depth must not be abandoned for areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain. The specific defense in depth required for these uncertainties is the uncredited pressure in the containment, which serves as a hedge for these uncertainties. The whole pressure in containment must be retained since the calculation methods are so uncertain. Giving up a portion of the containment pressure for overpressure credit for proper operation of ECCS pumps is an unacceptable erosion of the defense in depth provided by the pressure in containment. Without retention of the whole amount of pressure in containment for defense in depth, the uncertainties in the NPSH calculations dictate that it cannot be determined that reasonable assurance exists that public health and safety will be protected.

Third Contention

Because Applicant Is Voluntarily Seeking A Change In Design Or Licensing Basis, It Should Comply With Current, More Restrictive Practices Which Relate to the Proposed Design or Licensing Basis Change in Order to Demonstrate That it Will Provide Adequate Protection to the Health and Safety of the Public As Required By 42 U.S.C. §2232(a).

Bases

1. Taking credit for containment overpressure in order to meet NPSH requirements for ECCS pumps involves a change to the design or licensing basis for the plant.

2. When such changes are made voluntarily, as is the case here, the Applicant should then meet current more restrictive practices with regard to issues related to the proposed design or licensing basis change because the justification for "grandfathering" the plant as to such design or licensing basis changes no longer exists.

3. There are two issues which are directly related to the proposal to take credit for containment overpressure in order to meet NPSH requirements for ECCS pumps for which Applicant has not used the current more restrictive practices in its analysis:

a. Applicant has not evaluated the containment and its appurtenances under the current rules for single failure.

b. Applicant has not evaluated the proposed uprate in light of current assumptions for simultaneous safe shutdown earthquake (SSE) but relies on analytical methods and SSE values that have evolved dramatically.

Supporting Evidence

1. The Applicant's request for credit for containment overpressure is a request for a change in it's design or licensing basis (these two terms are used synonymously in this motion).

2. The Applicant wishes to implement this design basis change, which results from a change in NRC policy and practice, albeit improperly implemented, by using analyses related to the use of the reactor containment that are less restrictive than those currently in use. The Applicant is not implementing more restrictive analyses, resulting from similar design basis changes to NRC policies and practices, that are related to the use of the reactor containment.

This practice by the Applicant of seeking to take advantage of one design basis change authorized by the NRC while ignoring the related, and more restrictive design basis changes, also authorized by the NRC, is known throughout the industry as "cherry-picking."

3. NRC has established a precedent for an acceptable approach to the problem of regulatory cherry-picking in Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000 (DPS Exhibit 22) :

> **5.1.4 Applicability of Prior Licensing Basis**
>
> The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis.

With this approach, NRC staff may apply more restrictive, current practices to issues related to changes in design bases that are voluntarily initiated by the licensee.

4. In order to prevent regulatory cherry-picking in conjunction with the Applicant's power uprate application, two considerations associated with the request for a change in design basis related to containment overpressure credit for ECCS pumps are warranted.

5. Since the Applicant voluntarily wishes to use the reactor containment for a new design

basis function of maintaining a minimum level of pressure for up to 50 hours after an event, the Applicant needs to evaluate the containment and its appurtenances under the current rules for single failure. The Applicant's current design basis only assumes a single failure of active equipment or components. Current criteria requires assumption of a single active failure in the short term, or either a single active or passive failure in the long term. Current criteria considers check valve movement and spurious valve movement as single active failures and also considers the effects of a single inappropriate operator action. Applicant's analysis did not consider these as single active failures. Applicant has not evaluated the containment and its appurtenances to these current single failure criteria, and thus there is not reasonable assurance that the proposed crediting of containment overpressure will protect public health and safety.

6. The other area in which current practices must be applied is the seismic analysis of the reactor containment. The voluntary change in design basis for containment overpressure credit is requested in part for LOCA's. The Applicant's current design basis for LOCA's includes the assumption of simultaneous safe shutdown earthquake (SSE). However, the Applicant's design basis value for an SSE is only 0.14 g, and the analytical methods used by the Applicant have evolved dramatically. Newer nuclear plants in the New England region, Seabrook and Millstone 3, have significantly higher SSE accelerations than the Applicant.

7. The NRC summarized the evolution of seismic analysis methods as follows:

> Over the years, there has been an evolution of seismic design requirements and technology. Early nuclear power plants were designed without specific seismic design requirements. In the early 1970s, the requirement for resistance to seismic events was included in the regulations. The state of

> knowledge has advanced rapidly and the methods of seismic design vary with the vintage of the nuclear power plant. Also, the complex process of seismic design and analysis involved many engineering disciplines: seismic, geotechnical, structural, mechanical, electrical, and nuclear.

NUREG-0093. Item A-40 (DPS Exhibit 23).

8. The Vermont State Geologist also questions the adequacy of the Applicant's containment seismic analysis. He identifies aspects of current seismic analysis that appear more restrictive than the Applicant's analysis. See Vermont State Geologist letter of August 26, 2004, *Probability of Earthquake Induced Ground Accelerations at Vermont Yankee* (DPS Exhibit 24).

9. Containment isolation valves have frequently exceeded allowables in leakage tests. The Applicant has not demonstrated, from the as-found condition of containment isolation valves, that these valves will satisfactorily retain containment pressure for a period up to 50 hours following an earthquake using current seismic analysis standards.

10. If the containment does not adequately withstand an earthquake, the containment or its attached isolation valves could fail in a manner not to retain pressure. In this event, the containment overpressure would not be present for ECCS pump adequacy, and there could be a high likelihood that the ECCS pumps would fail, in turn causing fuel failure and fission product release.

11. Under current operation, we accept the adequacy of Vermont Yankee's current seismic analysis. However, for the new use of containment and voluntary design basis change, the containment must be analyzed to current seismic analysis method to demonstrate adequacy. Lacking the evaluation of the containment and its appurtenances to current seismic analysis

methods, there will not be reasonable assurance that the proposed crediting of containment overpressure will protect public health and safety.

Fourth Contention

The Change in Design Basis to Use the Reactor Containment as an Engineered Safety Feature to Guarantee at Least a Minimum Pressure for ECCS Pump Performance Violates the Lessons- Learned Regarding Human Factors for Operators in the Three Mile Island Event and Creates Contrary and Confusing Operating Requirements That Will Create a Significant Hazard (10 C.F.R. §50.92) and Will Not Provide Adequate Protection for the Public Health and Safety as Required by 10 C.F.R. §50.57(a)(3).

Bases

1. The primary and desired response by plant operators in an event which increases containment pressure is to reduce containment pressure. With the proposed design basis change to credit set levels of containment overpressure, the operators will be placed in the confused position of both needing to reduce containment pressure and to maintain containment pressure.

2. The Applicant's proposal related to emergency operator procedure would create the same unacceptable human factors paradigm for operators that was found by the Task Force which investigated the causes of the Three Mile Island, Unit 2, accident.

Supporting Evidence

1. The review of the Three Mile Island, Unit 2, accident revealed that human factors for plant operators and emergency operating procedures were a primary contributor.

> The principal conclusion of the Task Force is that, although the accident at Three Mile Island stemmed from many sources, the most important

> lessons learned fall in a general area we have chosen to call operational safety. This general area includes topics of human factors engineering, qualification and training of operations personnel; integration of the human element in the design, operation, and regulation of system safety; and quality assurance of operations. Specifically, the primary deficiency in the reactor safety technology identified by the accident was the inadequate attention that had been paid by all levels and all segments of the technology to the human element and its fundamental role in both the prevention of accidents and the response to accidents.

NUREG-0585, *TMI-2 Lessons Learned Task Force Final Report*, October 1979, at p. 1-2 (DPS Exhibit 25).

> The NRC [at the time of the TMI-2 accident] gives short shrift in the design safety review process to determining how well operators will be able to diagnose abnormal events, based on what they see on their instruments, and respond to them.

NUREG/CR-1250, Vol. 1, *Three Mile Island, A Report to the Commissioners and the Public*, NRC Special Inquiry Group, Mitchell Rogovin, Director, circa. 1980, at 122 (DPS Exhibit 26).

> The use of properly prepared procedures in plant operations is another important ingredient in the matrix of operational safety . . . Emergency operating procedures should consider system interactions and be written in such a manner that they are unambiguous and useful in crisis control . . . The Task Force has found the NRC review process for emergency procedures to be inadequate . . . Past practice was not sufficient because it did not specifically investigate the compatibility of emergency procedures with the design bases of the systems involved, nor was the discipline of human factors involved.

NUREG-0585, at p. 2-6.

> Emergency operating procedures for all nuclear power plants should be reviewed by the NRC. The review should be conducted by interdisciplinary review groups comprising I&E inspectors and NRR technical reviewers knowledgeable in system design, accident analysis, operator training, theories of education and crisis management, human factors, and the underlying technical bases for licensing.

Id., at p. A-9.

2. The use of reactor containment as an engineered safety feature to guarantee at least a minimum pressure for ECCS pump performance creates confusion for operators. Operators are trained, and have been trained for the past 32 years at Vermont Yankee, to take action to reduce containment pressure if it increases (for any reason) a small amount over atmospheric pressure. If the containment overpressure credit were granted, these operators would be required not only to concentrate on reducing containment pressure, but would also be required to retain a minimum amount of pressure.

3. The minimum pressure to retain is confusing since it is not a constant amount, but rather varies for different time steps, at times when operators would be diverted with many other contravening tasks to mitigate the various event. For example, the pressure credited for a LOCA includes these pressure steps over a 50 hour period: 2.4 psig, 3.4 psig, 4.4 psig, 5.1 psig, 6.1 psig, 5.6 psig, 5.1 psig, 4.6 psig, 4.1 psig, 3.6 psig, 3.1 psig, 2.6 psig, 2.1 psig, 1.7 psig, and 1.3 psig. VYC-0808, Rev. 6 (DPS Exhibit 16). Instead, if it is an ATWS, the pressure credited is 2.4 psig over a period of almost 2 hours. VYC-0808, Rev. 6, Change 4 (DPS Exhibit 27). If it is an SBO, the pressure credited varies from 0.5 psig to 2.1 psig over a period of almost three and one-half hours which begins six hours after the station loses power. VYC-2314, Rev. 0 (DPS Exhibit 28). Finally, if it is an Appendix R fire, pressure credited varies from 0.5 psig to 0.9 psig over a three and one-half hour period. VYC-2314, Rev. 0. This pressure crediting scheme is complicated for operators to grasp in the middle of emergencies.

4. It is highly undesirable to allow the containment pressure to be higher than necessary, because higher pressure would result in greater fission product leakage in a fission product release accident. It is not clear that operators will be able to control pressure within the limits required by the new proposed design basis. For example, the Applicant proposes to credit containment pressure following a LOCA at 6.1 psig from time 9000 seconds (2.5 hours) to time 400000 seconds (11.1 hours), a period of almost nine hours. If the operator uses maximum containment sprays, where should the pressure be stopped to keep 6.1 psig for nine hours? What will the operator do if he undershoots the credited pressure, or if the pressure drops over the nine hours below the 6.1 credit? These requirements create unacceptable levels of confusion for the operator and create the kind of situation described by the reviews of the Three Mile Island accident, quoted earlier.

5. Review of VYC-0808, Revision 6, Change 6, page 12 of 14 (Table 4.2 LOCA) (DPS Exhibit 21) identifies that for much of the 50 hour period that the Applicant proposes to credit overpressure, the difference between overpressure available and overpressure credited is between 1 psig and 1.5 psig for much of the time. This is too small a band for an operator to be able to control in the midst of a crisis with such dire consequences - the potential failure of ECCS cooling pumps.

6. The Applicant responded to an RAI on emergency operating procedures. The RAI illustrates that the Applicant, if allowed, would create the same type of unacceptable situation regarding emergency operating procedures described by the Three Mile Island accident Task

Force. The entire RAI and its response are repeated below:

RAI SPSB-C-22

Describe how the VYNPS emergency operating procedures will be revised to ensure that the containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH.

Response to RAI SPSB-C-22

The VYNPS emergency operating procedures (EOPs) do not require revision to ensure that the containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH. Current EOPs incorporate guidance to ensure that containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH.

Per VYNPS emergency operating procedure (EOP) EOP-1, "RPV Control," after an automatic action level has been reached, operators are directed to verify applicable automatic actions have occurred. Verifying automatic actions provides backup confirmation that all isolation valves have closed on a primary containment isolation signal.

VYNPS EOPs establish NPSH limits for residual heat removal (RHR) and core spray (CS) pumps. (Separate limits are provided for RHR and CS). The NPSH limit is a function of pump flow, torus water temperature, and suppression chamber pressure. It is used to preclude ECCS pump damage due to cavitation and to ensure adequate coolant flow. As overpressure increases, the static pressure and margin to saturation at the pump inlet also increase. The available NPSH therefore increases with overpressure.

In accordance with EOP-1, when using RHR for an injection system, operators are directed to inject through the heat exchanger as soon as possible and to control and maintain pump flow below the RHR NPSH Limit. For the core spray system, operators are directed to control and maintain pump flow below the CS NPSH Limit.

EOP-3, "Primary Containment Control," Note 5 states: "**Reducing primary containment pressure will reduce the available NPSH for pumps taking suction from the torus.**" Per the EOP Study Guide, if there is no future need for sprays and containment overpressure is desired to provide adequate NPSH for

> pumps drawing suction from the suppression pool, sprays may be terminated at a higher pressure.
>
> In accordance with EOP-3, drywell sprays are initiated before containment temperature reaches 280 IF or when torus pressure exceeds 10 psi. **Containment sprays should isolate automatically when drywell pressure decreases to 2.5 psig.** Both of these steps in EOP-3 provide reference to Caution #5 emphasizing the relationship between primary containment pressure and available NPSH.
>
> Also, per EOP-3, once the high drywell pressure isolation occurs, containment venting is directed only after a reactor pressure vessel emergency depressurization (RPV-ED) is required and prior to exceeding the primary containment pressure limit (PCPL-A curve in EOP-3). **In the event that containment venting is required, operators will vent the containment to control pressure below the PCPL-A curve.** The pressure at which containment is maintained during venting is based on considerations of NPSH for the RHR and core spray pumps, expected release rates, and total releases. Therefore, sufficient containment overpressure is preserved.

Applicant request for Extended Power Uprate, Supplement 8, Attachment 2, at 178-9 (Emphasis added) (DPS Exhibit 29).

7. The following are areas in which the Applicant's plans for emergency operator procedures create the same type of unacceptable situation described by the Three Mile Island Task Force:

- It is unacceptable that the Applicant does not plan to change EOPs to incorporate the new proposed design basis of credited overpressure. This means that while the Applicant proposes to license its designed based on this pressure, it will not have its operators attempt to maintain that pressure in accidents. Neither will the Applicant train operators to maintain the credited overpressure. The Task Force found "emergency operating procedures should . . . be written in such a manner

that they are unambiguous . . . Past practice was not sufficient because it did not specifically investigate the compatibility of emergency procedures with the design bases of the systems involved."

- The Applicant's note, "Reducing primary containment pressure will reduce the available NPSH for pumps taking suction from the torus," is unacceptable because it does not tell the operator he must maintain a set level of overpressure according to the licensing basis.
- The fact that containment sprays automatically terminate at 2.5 psig creates an additional step the operator must take during a crisis. This is inconsistent with the proposed licensing basis, which is to maintain overpressure at a range of pressures. On the one hand, to try to control to these licensing basis pressures will create great operator distraction. However, the Applicant's plan not to have the operator control to the licensing basis overpressure is a violation of that licensing basis. This fact illustrates the confusion created by the Applicant's proposal, and shows that overpressure credit should not be granted.
- The EOP's identify the possibility of containment venting. The possibilities of over venting or not being able to re-close the vent have not been investigated properly, and when investigated, will illustrate that overpressure credit should not be granted.
- The fact that EOP's have not been modified and cannot be reviewed by the NRC

staff is not acceptable. NRC review of EOP's was a cited weakness and contributing cause to the Three Mile Island accident. The NRC staff has accepted the TMI Task Force recommendation and has devoted much interdisciplinary review to EOP's. However, the incorporation of this proposed change in design basis related to containment overpressure should receive the same level of interdisciplinary review as the EOP's on the whole. It is unacceptable that the Applicant is creating a situation in which the NRC staff will not give the changes to the EOP's the necessary interdisciplinary review.

Fifth Contention

To the Extent Applicant Is Claiming That Use of Containment Overpressure as a Credit to Meet NPSH Is Necessary and Failure to Use it Is Impracticable Because of Economic or Need for Power Considerations, its Request Should Be Rejected as Contrary to the Atomic Energy Act (42 U.S.C. §2232).

Bases

1. Regulatory Guide 1.82, Revision 3, authorizes the use of containment overpressure to meet NPSH requires when it is "necessary" or when it would be "impracticable" to alter the plant to meet NPSH requirements. The normal meaning of these terms implicates economic considerations.

2. Applicant has not demonstrated that there is no available alternative to use of containment overpressure to meet NPSH requirement and in fact either lowering the level of the

proposed uprate or upgrading the ECCS pumps would allow Vermont Yankee to meet NPSH requirements.

3. It is well-established under the Atomic Energy Act by decisions of federal courts and the Commission, that cost considerations are irrelevant to determining whether safety requirements have been met.

4. The Applicant cannot excuse failure to meet NPSH requirements without the use of containment overpressure by asserting that meeting such requirements, without the use of containment overpressure, is too expensive or will reduce power output below the proposed 20% uprate.

Supporting Evidence

1. The evidence related to the technical issues raised by this contention is contained in the Supporting Evidence related to the First through the Fourth Contentions.

2. The legal evidence in support of this Contention includes the following:

- In setting or enforcing the standard of "adequate protection" that this section [42 U.S.C. §2232] requires, the Commission may not consider the economic costs of safety measures. The Commission must determine, regardless of costs, the precautionary measures necessary to provide adequate protection to the public; the Commission then must impose those measures, again regardless of costs, on all holders of or applicants for operating licenses.

  Union of Concerned Scientists v. U.S. Nuclear Regulatory Com'n 824 F.2d 108, 114, (C.A.D.C.,1987)

- *Power Reactor Development Co. v. International Union of Electrical, Radio, and Machine Workers,* 367 U.S. 396, 81 S.Ct. 1529, 6 L.Ed.2d 924 (1961).

- *Maine Yankee Atomic Power Co.,* 6 A.E.C. 1003 (1973).

B. HEARING ON THESE CONTENTIONS SHOULD BE CONDUCTED PURSUANT TO THE PROCEDURES IN 10 CFR PART 2, SUBPART G

In adopting the current Rules of Practice the NRC noted the following:

> The AEC of the 1950s asserted that formal hearings were required by Section 189.a. At that time, the AEC saw benefits in a highly formal process, resembling a judicial trial, for deciding applications to construct and operate nuclear power plants. It was thought that the panoply of features attending a trial—parties, sworn testimony, and cross-examination—would lead to a more satisfactory resolution of the complex issues affecting the public health and safety and would build public confidence in the AEC's decisions and thus in the safety of nuclear power plants licensed by the AEC.

69 F.R. 2182, 2183 (January 14, 2004). Although the NRC has now determined that these principles are no longer universally relevant to its hearings, DPS respectfully submits that these principles are very relevant to the contentions it raises in this particular hearing. Vermont has demonstrated a keen and continuing interest in its one nuclear power plant. Elected officials, including the Governor and the entire Congressional delegation, have already expressed their concern that adequate time be provided to prepare for this hearing to fully explore the many complex issues presented by Applicant's proposal to essentially add 100MW of nuclear power to Vermont's generating capacity. DPS letter to NRC, December 8, 2003; DPS letter to NRC, June 8, 2004 (DPS Exhibits 13, 19), NRC Order denying request for delay of deadline to file for

hearing, August 18, 2004. DPS has been actively involved in oversight of Vermont Yankee ever since the plant received its operating license and has, through the efforts of its staff nuclear engineer, William K. Sherman, maintained a physical presence at Vermont Yankee at crucial times and during periodic reviews. Most recently, the NRC, at the request of Governor Douglas and the Vermont Public Service Board, is conducting an independent engineering inspection directly related to the complex safety issues raised by this Application for a 20% power uprate. Mr. Sherman fully participated in that review on behalf of the DPS and the State as an observer. Many Vermonters are very interested in and concerned about the proposal to increase the level of nuclear power output from the State's only commercial nuclear power plant.

In light of these considerations, we believe it is essential that the NRC have a full hearing, with live witnesses and cross-examination of those witnesses, and full discovery with document production requests and depositions, to assure the public that whatever decision is reached, there has been a full and public airing of the important safety issues which this proposal raises. The fact that the issues involved are extremely complex underscores the need for such a public hearing where witnesses, compelled to address a hearing board composed of at least some members who are not nuclear engineers, will be required to put into understandable terms their concerns about the proposed uprate and the answers to those concerns. The above articulation of the Contentions which DPS believes should be addressed and the bases and supporting evidence for those conditions provide ample evidence that the issues involved in this Application are neither trivial nor simple. They are concerned with the safety policies of the NRC, the work of

the ACRS and research conducted by nuclear engineers at national laboratories and research centers across the country.

10 CFR §2.310(d) provides that a hearing "will be conducted under subpart G" if, *inter alia*, the presiding officer "finds that resolution of the contention or contested matter necessitates resolution of issues of material fact relating to the occurrence of a past activity". *Id.* As will be evident when the Applicant files its response to this Petition to Intervene, there is substantial controversy, both regarding the facts and the interpretation to be placed on those facts as these relate to past activities. Among the issues which we expect will require resolution by the Board of material fact disagreements related to past activities are 1) how did Applicant calculate post-accident conditions in making its determination of the level of post-accident containment pressure and was this calculation appropriate?, 2) did testing conducted of the performance of ECCS pumps following a LOCA leave a large area of uncertainty regarding NPSH including the impact of strainers and debris on NPSH?, 3) did the ACRS actually conduct the statutorily required safety review of the portion of Regulatory Guide 1.82, Revision 3, which altered the long-standing NRC prohibition against using containment overpressure as a credit to meet NPSH for ECCS pumps following a LOCA?, 4) does defense in depth as traditionally developed by the NRC and used in licensing decisions prohibit allowing failure of one physical barrier, in this case the reactor containment, to result in the failure of the ECCS pump function which in term will fail a second physical barrier, the fuel cladding, and if so is the level of uncertainty associated with the calculation of post-LOCA NPSH and containment performance sufficiently high to

make reliance on probabilistic risk analyses (PRA) instead of defense in depth, unacceptable? and 5) has Applicant provided sufficient evidence to prove that meeting NPSH requirements without taking credit for containment overpressure by altering the plant or the proposed level of uprate is "impracticable" or that use of containment overpressure is necessary? Should the Staff decide to participate they will add further controversy to these issues.

These are not issues which can be rationally decided on the bare bones of the written word. When such complex and controversial issues are involved, oral presentations, with the benefit of probing questions from the parties and the Board are the only way to get to the facts.

> We distinguish between the assertion of a broad right of cross-examination, such as that argued to this court, and a claim of a need for cross-examination of live witnesses on a subject of critical importance which could not be adequately ventilated under the general procedures. This is the kind of distinction that this court made in its en banc opinion in American Airlines v. CAB, *supra*, 123 U.S.App.D.C. at 318-319, 359 F.2d at 632-633. We see no principled manner in which firm time limits can be scheduled for cross-examination consistent with its unique potential as an "engine of truth"-the capacity given a diligent and resourceful counsel to expose subdued premises, to pursue evasive witnesses, to "explore" the whole witness, often traveling unexpected avenues.

International Harvester Co. v. Ruckelshaus 478 F.2d 615, 631 (C.A.D.C.1973). Where issues of the complexity involved in this proceeding are presented it is unrealistic to expect that the parties can fully develop their issues without being able to ask and receive answers to their questions or that the Board can resolve disagreements among the parties about the facts and the interpretations to be placed on those facts without the benefit of live testimony to "expose subdued premises . . .

and to 'explore' the whole witness, often traveling unexpected avenues". *Id.*

Similarly, the complexity of the issues and the far reaching nature of the documents which may shed light on these issues, including the actual tests run and analyses performed to determine the level of risk associated with the post accident impacts on ECCS pump operation and the underlying documentation which is alleged to support Applicant's conclusions regarding the containment pressure following a postulated-LOCA, warrant allowing an opportunity for full document production requests which can obtain information beyond the hearing file and beyond the information voluntarily produced by the Applicant pursuant to 10 CFR §2.336. The use of depositions will have the salutary effect of reducing the hearing time and will allow a fuller opportunity for the witness to make his/her position clear and for the examiner to probe all the bases for those positions. DPS has a number of concerns which might be satisfied by full discovery and which might actually reduce the number of issues to be raised at the hearing. DPS has neither an interest in or motive for using the discovery process for any purpose other than getting at the correct statement of the facts. The record as it now stands makes it impossible to determine whether the Applicant has stated the facts correctly and without full discovery and cross-examination rights, DPS respectfully submits the Board will not be able to determine the correct statement of the facts. Deficiencies in the record and uncertainties over critical issues ultimately disadvantages the party with the burden of proof. In this case that is the Applicant. DPS believes the public will be ill-served by rejecting the Application on the basis of an incomplete record just as it would be ill-served by the granting the Application when important

safety issues remain unresolved.

Listed below are a few examples of the document production requests DPS would make and the areas of examination which DPS would explore in deposition or at the hearings.

Please identify all initial conditions, inputs, and assumptions for analysis for the following:

- Determination of torus temperature for LOCA, SBO, ATWS, and Appendix R fire events
- Determination of available NPSH for LOCA, SBO, ATWS, and Appendix R fire events
- Determination of head losses for piping, clean strainer, and debris loading for LOCA, SBO, ATWS and Appendix R fire events

Please provide the Vermont Yankee Calculations used for the determinations identified above.

Please provide copies of all references from the calculations provided above.

Please provide a copy of all emergency operating procedures.

Please provide copies of operating procedures applicable for LOCA, SBO, ATWS and Appendix R fire events, and related actions and references.

Please provide copies of all training material for operators regarding the assuring the adequate performance of the residual heat removal and core spray pumps during LOCA, SBO, ATWS, and Appendix R fire events.

Please identify and provide documentation for all tests run to determine NPSH and to verify the adequacy of NPSH of the residual heat removal and core spray pumps.

C. RESERVATION OF RIGHT TO AMEND

In furtherance of its interests in the Vermont Yankee extended power uprate, the State, by requests from Governor Douglas and the Vermont Public Service Board, requested the NRC to conduct an independent and in-depth review of a number of important features at Vermont Yankee. The NRC agreed to conduct a significant portion of the review sought by the State. In agreeing to conduct an independent inspection, Chairman Diaz described the process to be used:

> Over the past several months, the NRC has been developing a new engineering inspection program which we intend to pilot at selected plants. The NRC staff considered a number of factors, including the Board's request for an independent engineering assessment, and concluded it is appropriate to conduct this engineering inspection at Vermont Yankee. This new engineering assessment inspection incorporates the best practices of the existing and past engineering inspections. The NRC will use this inspection to verify that design bases have been correctly implemented for a sampling of components across multiple systems and to identify latent design issues. The inspection process uses operating experience, risk assessment, and engineering analysis to select risk significant components and operator actions, and will ensure that adequate safety margins exist. Although the specific sampling of components is still being developed, it will include components from multiple systems that are potentially affected by a power uprate such as the emergency core cooling systems, the containment system, power conversion systems, and auxiliary systems.

Letter, Nils J. Diaz to Michael H. Dworkin (5/4/04). Among the issues to be investigated are "changes that could impact the integrity of barriers (e.g., higher flow rates which could increase vibration at specific support points), safety evaluations, plant modifications, post maintenance and surveillance testing, heat exchanger performance, and integrated plant operation." Established NRC Power Uprate Review Process with letter from Diaz to Dworkin (5/4/04).

DPS believes completion of this inspection, now scheduled for mid-September 2004, will

provide critical information relevant to issues which are likely to require thorough evaluation in the NRC hearing process, including some issues already identified as to which the review may provide relevant information and bases for modified contentions or elimination of a contention. Attempting at this time, without the benefit of the results of the review, to identify all the appropriate issues, provide the bases for each issue, identify supporting information for each of those bases and demonstrate how resolution of those issues requires a full adjudicatory hearing, is not feasible. Thus, motions to amend the filed contentions are almost certain to be filed. If action is taken on the now filed contentions before proposed amended contentions, bases and/or supporting evidence are submitted, it is likely the Board will waste its own time and the time of the parties and potential parties. On the other hand, by delaying the date for action on the requests for intervention until 30 days after the full report of the independent inspection and its supporting documents have been made publicly available, will enable the parties to better identify any issues which require resolution, the bases for these issues, the information which supports these issues and the reason why an adjudicatory hearing is required.

Although 10 CFR §§2.309(c) and (f)(i)(ii) and (iii) provide narrow opportunities for submitting new contentions or amending previous contentions, each imposes additional hurdles which are not applicable to initial contentions. Moreover the use of such procedures following the issuance of the independent engineering inspection, will ultimately delay the hearing process and enmesh the Board or Commission and the parties in an unnecessary wrangle over the application of a procedural rule rather than maintaining focus on the substantive issues involved

in the uprate proceeding. It makes far more sense for the Board to allow amendments to the contentions, bases and supporting evidence and the request for adjudicatory hearing to be filed within 30 days of the public availability of independent engineering inspection report and supporting documentation without the constraints imposed by 10 CFR §§2.309(c) and (f)(i)(ii) and (iii). Otherwise the Board will have devoted considerable time first to determining the intervention status of the parties based on filings made on August 30th and then will have to reconsider those decisions in light of the new submittals based on the independent engineering inspection report as well as determining whether the procedural requirements of 10 CFR §§2.309(c) and (f)(i)(ii) have been met. Inasmuch as the independent engineering inspection report is scheduled for release in mid-September and by application of the procedures of 10 CFR §2.309(h) the middle of October is the earliest the Board could begin to consider the Petitions, responses and replies, there is virtually no time lost by allowing amendments of contentions, bases and supporting evidence and requests for adjudicatory hearings to be made within 30 days after the independent engineering inspection report and its documentation are made public.

CONCLUSION

For all the reasons stated, the State of Vermont, acting through its Department of Public Service requests that an adjudicatory evidentiary hearing under 10 CFR Part 2, Subpart G be held to fully examine the contentions it has raised in this pleading and any subsequent amendments it may submit to these contentions.

Respectfully submitted,

Sarah Hofmann
Special Counsel
Department of Public Service
112 State Street - Drawer 20
Montpelier, VT 05620-2601

Anthony Z. Roisman
National Legal Scholars Law Firm
84 East Thetford Rd.
Lyme, NH 03768

August 30, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In Re: Entergy Nuclear Vermont Yankee)
LLC and Entergy Nuclear) **Docket No. 50-271**
Operations, Inc.) **(Extended Power Uprate)**

AFFIDAVIT OF WILLIAM K. SHERMAN

1. My name is William K. Sherman. I am employed by the Vermont Public Service Department in the position of State Nuclear Engineer. I have held this position since November, 1988. My duties include ongoing State regulatory oversight of the Vermont Yankee Nuclear Power Station ("Vermont Yankee"), as well as advising the Department and other State agencies on issues related to Vermont Yankee and nuclear power. My resume is attached to this affidavit.

2. I assisted in the preparation of the Vermont Department of Public Service Notice of Intention to Participate and Petition to Intervene ("VDPS Petition").

3. All of the information given as Supporting Evidence in Contentions 1 through 4 of the VDPS Petition is true and correct to the best of my knowledge.

William K. Sherman
William K. Sherman
State Nuclear Engineer

Subscribed and sworn to before me this 29th day of August, 2004.

David J. Mullett
David J. Mullett
Notary Public
My commission expires February 10, 2007

William K. Sherman

Mr. Sherman has a broad range of policy, public relations, economic and technical experience in the nuclear area over a thirty five-year career.

Professional Employment

1988 - Present	**Vermont Department of Public Service** State Nuclear Engineer
1973 -1985	**Stone & Webster Engineering Corporation** Senior Power Engineer
1971 - 1973	**EDS Nuclear, Inc.** Senior Engineer
1967 - 1971	**U.S. Naval Nuclear Power Program** Lieutenant

Experience

Vermont Department of Public Service

Cognizance of the daily status of operation of the Vermont Yankee Nuclear Plant.

Periodic inspections at the Vermont Yankee Nuclear Plant.

Liaison with the federal regulator of the Vermont Yankee Nuclear Plant.

Responsibility for monitoring and evaluating physical plant conditions during nuclear emergencies.

Maintains cognizance of issues and activities related to nuclear power in support of the Commissioner's position as NRC State Liaison Officer.

Expert witness testimony for the Department for issues associated with Vermont Yankee and nuclear power.

Serves as Vermont's Member on the Texas Low-level Radioactive Waste Disposal Compact Commission.

Serves as a member of the Nuclear Waste Strategy Coalition, a coalition of state public utility commission, attorney general and nuclear utility representatives, acting to effect a solution for the disposal of nuclear high-level radioactive waste.

Serves as a member and past-chairman of the Northeast High-Level Radioactive Waste Transportation Task Force.

Testifies before legislative committees on nuclear power issues.

Serves as principal staff for the Vermont State Nuclear Advisory Panel (VSNAP).

Experience - (continued)

Stone & Webster Engineering Corporation

Environmental Qualification Manager for a nuclear power plant under construction (May 1985 - Jan 1986). Supervised compliance with the requirements for environmental qualification of Class 1E electrical equipment.

Lead Power Engineer (Mar 1982 - May 1985) for a nuclear power plant under construction. Responsible for the overall technical and administrative direction of the power-related engineering and design activities associated with the 1200 MW pressurized water reactor in the construction phase. Direction of ongoing efforts such as preparation of System Descriptions and the Final Safety Analysis Report.

Principal Nuclear Engineer (Feb 1981 - Apr 1982) for a nuclear power plant under construction. Responsible for nuclear-related engineering and design activities during the construction phase. Supervised the activities of Engineers responsible for the NSSS contract, nuclear systems, nuclear-related buildings, and major specifications.

Power Engineer, assigned to the Nuclear Engineering Group (Feb 1980 - Feb 1981) for a nuclear power plant under construction. Coordinated all activities for the fuel building and fuel handling systems, and for the auxiliary building and component cooling water system. Responsible for safety-related specifications for pumps, heat exchangers, and cranes.

Lead Licensing Engineer (Mar 1973 - Jan 1980). Responsible for project activities toward obtaining construction permits for three nuclear projects. Supervised the preparation of the Safety Analysis Reports and Environmental Reports. Responsible for evaluation of plant design to ensure compliance with NRC licensing requirements. Responsible for liaison with federal and state regulatory agencies.

EDS Nuclear, Inc.

Licensing and engineering consulting work for a number of nuclear utiities.

U.S. Naval Nuclear Power Program

Instructor at U.S. Naval Nucelar Power School in the areas of Reactor Physics, Heat Transfer, and Physics.

Education

1963 - 1967	**The University of Michigan** Bachelor of Science (Mechanical Engineering)

Licenses

Registered Professional Engineer - California, Massachusetts, Connecticut

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	Docket No. 50-271
Entergy Nuclear Vermont Yankee LLC	)	(Extended Power Uprate)
and Entergy Nuclear Operations, Inc.	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Vermont Department of Public Service Notice of Intention to Participate and Petition to Intervene have been served upon the following persons by U.S. Mail, first class or electronic mail as indicated.

VIA U.S. Mail:
Lawrence J. Chandler, Esq.
Office of the General Counsel
Mail Stop - O-15 D21
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

John M. Fulton, Esq.
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Jay Silberg, Esq.
Shaw Pittman LLP
2300 N. Street, NW
Washington DC 20037

Anthony Z. Roisman, Esq.
National Legal Scholars Law Firm
84 East Thetford Rd.
Lyme, NH 03768

VIA ELECTRONIC MAIL:
OGCMailCenter@nrc.gov

Sarah Hofmann, Special Counsel
Vermont Department of Public Service

Dated this 30th day of August, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION

In Re: Entergy Nuclear Vermont Yankee LLC and Entergy Nuclear Operations, Inc.	)))	Docket No. 50-271 (Extended Power Uprate)

EXHIBIT LIST TO VERMONT DEPARTMENT OF PUBLIC SERVICE NOTICE OF INTENTION TO PARTICIPATE AND PETITION TO INTERVENE

1. Draft general design criteria published July 11, 1967 (32 FR 10213)

2. NRC Regulatory Guide 1.82, Revision 3, *Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident*

3. Safety Guide (Regulatory Guide) 1.1

4. Regulatory Guide 1.82 (Rev. 0), June 1974, *Sumps for Emergency Core Cooling and Containment Spray Systems*

5. Unresolved Safety Issue (USI) A-43, *Containment Emergency Sump Performance*

6. NRC Bulletin 96-03, *Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors*

7. Generic Safety Issue (GSI) 191, *Assessment of Debris Accumulation on PWR Sump Pump Performance* (DPS Exhibit 7)

8. NRC Bulletin 2003-01, *Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors*

9. ACRS Thermal-Hydraulic Phenomena Subcommittee transcript, August 20, 2003

10. ACRS Full Committee transcript, September 11, 2003

11. ACRS letter of September 30, 2003, *Draft Final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-term Recirculation Cooling Following a Loss-of-coolant Accident."*

12. ACRS letter, May 19, 1999, *The Role of Defense in Depth in a Risk-informed Regulatory System*

13. DPS letter of December 8, 2003 to the NRC Staff

14. NRC June 29, 2004 letter to DPS, response to Dec 8 letter

15. Docket No. 6812, Prefiled Direct Testimony, DPS Witness William Sherman, May 9, 2003

16. Vermont Yankee Calculation VYC-0808, Rev. 6

17. RAI SPSB-C-25

18. Section 4.2.6 of Safety Analysis Report for Constant Pressure Power Uprate ("PUSAR")

19. DPS letter of June 8, 2004 to the NRC Staff

20. Vermont Yankee Calculation VYC-0808, Rev. 6, Change 5, July 1, 2004

21. Vermont Yankee Calculation VYC-0808, Rev. 6, Change 6, July 16, 2004

22. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000

23. Unresolved Safety Issue Item A-40: *Seismic Design Criteria*

24. Vermont State Geologist letter of August 26, 2004, *Probability of Earthquake Induced Ground Accelerations at Vermont Yankee*

25. NUREG-0585, *TMI-2 Lessons Learned Task Force Final Report*, October 1979

26. NUREG/CR-1250, Vol. 1, *Three Mile Island, A Report to the Commissioners and the Public*, NRC Special Inquiry Group, Mitchell Rogovin, Director, circa. 1980

27. Vermont Yankee Calculation VYC-0808, Rev. 6, Change 4

28. Vermont Yankee Calculation VYC-2314, Rev. 0

29. RAI SPSB-C-22

2.

PART 50 - LIC...SING OF PRODUCTION AND UTI...ZATION FACILITIES

General Design Criteria for Nuclear Power Plant Construction Permits

32 FR 10213
Published 7/11/67
Comment period
expires 9/9/67

The Atomic Energy Comission has under consideration an amendment to its regulation, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits."[1] The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria to be included in applications for Commission construction permits. These General Design Criteria would not add any new requirements, but are intended to describe more clearly present Commission requirements to assist applicants in preparing applications.

The proposed amendment would complement other proposed amendments to Part 50 which were published for public comment in the FEDERAL REGISTER on August 16, 1966 (31 F.R. 10891).

The proposed amendments to Part 50 reflect a recommendation made by a seven-member Regulatory Review Panel, appointed by the Commission to study: (1) The programs and procedures for the licensing and regulation of reactors and (2) the decision-making process in the Commission's regulatory program. The Panel's report recommended the development, particularly at the construction permit stage of a licensing proceeding, of design criteria for nuclear power plants. Work on the development of such criteria had been in process at the time of the Panel's study.

As a result, preliminary proposed criteria for the design of nuclear power plants were discussed with the Commission's Advisory Committee on Reactor Safeguards and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. In developing the proposed criteria set forth in the proposed amendments to Part 50, the Commission has taken into consideration comments and suggestions from the Advisory Committee on Reactor Safeguards, from members of industry, and from the public.

Section 50.34, paragraph (b), as published for comment in the FEDERAL REGISTER on August 16, 1966, would require that each application for a construction permit include a preliminary safety analysis report. The minimum information to be included in this preliminary safety analysis report is (1) a description and safety assessment of the site, (2) a summary description of the facility, (3) a preliminary design of the facility, (4) a preliminary safety analysis and evaluation of the facility, (5) an identification of subjects expected to be technical specifications, and (6) a preliminary plan for the organization, training, and operation. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;

The "General Design Criteria for Nuclear Power Plant Construction Permits" proposed to be included as Appendix A to this part are intended to aid the applicant in development item (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

Section 50.34 as published in the FEDERAL REGISTER on August 16, 1966, would be further amended by adding to Part 50 a new Appendix A containing the General Design Criteria applicable to the construction of nuclear power plants and by a specific reference to this Appendix in § 50.34, paragraph (b).

The Commission expects that the provisions of the proposed amendments relating to General Design Criteria for Nuclear Power Plant Construction Permits will be useful as interim guidance until such time as the Commission takes further action on them.

Pursuant to the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act of 1946, as amended, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

1. Section 50.34(b)(3)(i) of 10 CFR Part 50 is amended to read as follows:

§ 50.34 Contents of applications; technical information safety analysis report.[2]

* * * * *

(b) Each application for a construction permit shall include a preliminary safety analysis report. The report shall cover all pertinent subjects specified in paragraph (a) of this section as fully as available information permits. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility, including:

(i) The principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.

* * * * *

2. A new Appendix A is added to read as follows:

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS[1]

TABLE OF CONTENTS

[1] Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

[2] Inasmuch as the Commission has under consideration other amendments to § 50.34 (31 F.R. 10891), the amendment proposed herein would be a further revision of § 50.34 (b)(3)(i) previously published for comment in the FEDERAL REGISTER.

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[1] Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

Introduction. Every applicant for a construction permit is required by the provisions of § 50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for those in Category B.

I. OVERALL PLANT REQUIREMENTS

Criterion 1—Quality Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Criterion 2—Performance Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) Appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Criterion 3—Fire Protection (Category A). The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Criterion 4—Sharing of Systems (Category A). Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Criterion 5—Records Requirements (Category A). Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

Criterion 6—Reactor Core Design (Category A). The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Criterion 7—Supression of Power Oscillations (Category B). The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Criterion 8—Overall Power Coefficient (Category B). The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Criterion 9—Reactor Coolant Pressure Boundary (Category A). The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 10—Containment (Category A). Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. Nuclear and Radiation Controls

Criterion 11—Control Room (Category B). The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Criterion 12—Instrumentation and Control Systems (Category B). Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Criterion 13—Fission Process Monitors and Controls (Category B). Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Criterion 14—Core Protection Systems (Category B). Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Criterion 15—Engineered Safety Features Protection Systems (Category B). Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Criterion 16—Monitoring Reactor Coolant Pressure Boundary (Category B). Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Criterion 17—Monitoring Radioactivity Releases (Category B). Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Criterion 18—Monitoring Fuel and Waste Storage (Category B). Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. Reliability and Testability of Protection Systems

Criterion 19—Protection Systems Reliability (Catergory B). Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Criterion 20—Protection Systems Redundancy and Independence (Category B). Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Criterion 21—Single Failure Definition (Category B). Multiple failures resulting from a single event shall be treated as a single failure.

Criterion 22—Separation of Protection and Control Instrumentation Systems (Category B). Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Criterion 23—Protection Against Multiple Disability for Protection Systems (Category B). The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Criterion 24—Emergency Power for Protection Systems (Category B). In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Criterion 25—Demonstration of Functional Operability of Protection Systems (Category B). Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Criterion 26—Protection Systems Fail-Safe Design (Category B). The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. Reactivity Control

Criterion 27—Redundancy of Reactivity Control (Category A). At least two independent reactivity control systems, preferably of different principles, shall be provided.

Criterion 28—Reactivity Hot Shutdown Capability (Category A). At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Criterion 29—Reactivity Shutdown Capability (Category A). At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Criterion 30—Reactivity Holddown Capability (Category B). At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Criterion 31—Reactivity Control Systems Malfunction (Category B). The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Criterion 32—Maximum Reactivity Worth of Control Rods (Category A). Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. Reactor Coolant Pressure Boundary

Criterion 33—Reactor Coolant Pressure Boundary Capability (Category A). The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Criterion 34—Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A). The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Criterion 35—Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A). Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120° F. above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60° F. above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Criterion 36—Reactor Coolant Pressure Boundary Surveillance (Category A). Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. Engineered Safety Features

Criterion 37—Engineered Safety Features Basis for Design (Category A). Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features

shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Criterion 38—Reliability and Testability of Engineered Safety Features (Category A). All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Criterion 39—Emergency Power for Engineered Safety Features (Category A). Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Criterion 40—Missile Protection (Category A). Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Criterion 41—Engineered Safety Features Performance Capability (Category A). Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Criterion 42—Engineered Safety Features Components Capability (Category A). Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Criterion 43—Accident Aggravation Prevention (Category A). Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Criterion 44—Emergency Core Cooling Systems Capability (Category A). At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Criterion 45—Inspection of Emergency Core Cooling Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Criterion 46—Testing of Emergency Core Cooling Systems Components (Category A). Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 47—Testing of Emergency Core Cooling Systems (Category A). A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Criterion 48—Testing of Operational Sequence of Emergency Core Cooling Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Criterion 49—Containment Design Basis (Category A). The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Criterion 50—NDT Requirement for Containment Material (Category A). Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30° F. above nil ductility transition (NDT) temperature.

Criterion 51—Reactor Coolant Pressure Boundary Outside Containment (Category A). If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Criterion 52—Containment Heat Removal Systems (Category A). Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Criterion 53—Containment Isolation Valves (Category A). Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Criterion 54—Containment Leakage Rate Testing (Category A). Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Criterion 55—Containment Periodic Leakage Rate Testing (Category A). The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Criterion 56—Provisions for Testing of Penetrations (Category A). Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Criterion 57—Provisions for Testing of Isolation Valves (Category A). Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Criterion 58—Inspection of Containment Pressure-Reducing Systems (Category A). Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Criterion 59—Testing of Containment Pressure-Reducing Systems Components (Category A). The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 60—Testing of Containment Spray Systems (Category A). A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Criterion 61—Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A). A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Criterion 62—Inspection of Air Cleanup Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

Criterion 63—Testing of Air Cleanup Systems Components (Category A). Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Criterion 64—Testing of Air Cleanup Systems (Category A). A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Criterion 65—Testing of Operational Sequence of Air Cleanup Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. FUEL AND WASTE STORAGE SYSTEMS

Criterion 66—Prevention of Fuel Storage Criticality (Category B). Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Criterion 67—Fuel and Waste Storage Decay Heat (Category B). Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Criterion 68—Fuel and Waste Storage Radiation Shielding (Category B). Shielding for radiation protection shall be provided in the design of spent fuel and waste storage

facilities as required to meet the requirements of 10 CFR 20.

Criterion 69—Protection Against Radioactivity Release From Spent Fuel and Waste Storage (*Category B*). Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

Criterion 70—Control of Releases of Radioactivity to the Environment (*Category B*). The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Washington, D.C., this 28th day of June 1967.

For the Atomic Energy Commission.

W. B. McCOOL,
Secretary.

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WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT

A. INTRODUCTION

General Design Criterion (GDC) 35, "Emergency Core Cooling"; GDC 38, "Containment Heat Removal"; and GDC 41, "Containment Atmosphere Cleanup," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require that systems be provided to perform specific functions, i.e., emergency core cooling, containment heat removal, and containment atmosphere clean up following a postulated design basis accident. Pursuant to GDC 36, "Inspection of Emergency Core Cooling System"; GDC 39, "Inspection of Containment Heat Removal System"; and GDC 42, "Inspection of Containment Atmosphere Cleanup Systems" of Appendix A to 10 CFR Part 50, these systems must be designed to permit appropriate periodic inspection of important components. Pursuant to GDC 37, "Testing of Emergency Core Cooling System"; GDC 40, "Testing of Containment Heat Removal System"; and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" of Appendix A to 10 CFR Part 50, these systems must be designed to permit appropriate periodic testing to ensure their integrity and operability. In addition, GDC 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

This guide describes methods acceptable to the NRC staff for implementing these requirements with respect to the sumps and suppression pools performing the functions of water sources for emergency core cooling, containment heat removal, or containment atmosphere clean up. The guide also provides guidelines for evaluating the adequacy of the availability of the sump and suppression pool for long-term recirculation cooling following a loss-of-coolant accident (LOCA). This guide applies to light-water-cooled reactors.

Additional information is provided in NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors"; NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors"; NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode"; NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers"; Supplement 1 to NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers"; Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage"; and Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."

This regulatory guide has been revised to enhance the debris blockage evaluation guidance for pressurized water reactors. Research after the issuance of Revision 2 indicated that the previous guidance was not comprehensive enough to ensure adequate evaluation of a pressurized water reactor (PWR) plant's susceptibility to the detrimental effects caused by debris accumulation on debris interceptors (e.g., trash racks and sump screens). The sections pertaining to PWRs have been changed, and minor changes have been made to the sections on boiling water reactors (BWRs) to make them consistent with current staff positions as described in the Safety Evaluation on the Boiling Water Reactor Owners Group's (BWROG's) Utility Resolution Guide (URG) for ECCS Suction Strainer Blockage (1998).

This regulatory guide has also been revised to include guidance previously provided in Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps." The provisions of Regulatory Guide 1.1 have been updated in this guide to reflect the results of the NRC's review of responses to Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

GENERAL

The primary safety concerns regarding long-term recirculation cooling following a LOCA are (1) LOCA-generated and pre-LOCA debris materials transported to the debris interceptors (i.e., trash racks, debris screens, suction strainers) resulting in adverse blockage effects, (2) post-LOCA hydraulic effects, particularly air ingestion, and (3) the combined effects of items (1) and (2) on long-term recirculation pumping operability (i.e., effect on net positive suction head (NPSH) available at the pump inlet). These emergency core cooling system (ECCS) safety concerns extend to the containment spray systems (CSS) for plants with containment designs in which the containment spray systems draw suction from the recirculation sump. In some cases, the containment spray systems would draw from the recirculation sump significantly earlier than would the ECCS.

Debris resulting from a LOCA, together with debris that exists before a LOCA, could block the emergency core cooling (ECC) debris interceptors and result in degradation or loss of NPSH margin. Such debris can be divided into the following categories: (1) debris that is generated by the LOCA and is transported by blowdown forces (e.g., insulation, paint), (2) debris that is generated or transported by washdown, and (3) other debris that existed before a LOCA (e.g., corrosion material, sludge in a BWR suppression pool) and that may become suspended in the containment sump or suppression pool. Debris can be further subdivided into (1) debris that has a high density and could sink but is still subject to fluid transport if local recirculation flow velocities are high enough, (2) debris that has an effective specific gravity near 1.0 and tends to remain suspended or sink slowly and will nonetheless be transported by very low velocities or local fluid turbulence phenomena, and (3) debris that will float indefinitely by virtue of low density and will be transported to and possibly through the debris interceptors. Debris generation, early debris transport, long-term debris transport, and attendant blockage of debris interceptors should be evaluated to ensure that the ability of the ECCS to provide long-term post-LOCA core cooling is not jeopardized. All potential debris sources should be evaluated, including but not limited to, the fire barrier material, insulation materials (e.g., fibrous, ceramic, and metallic), filters, corrosion material, and paints or coatings. Relevant information for such evaluations is provided in the Regulatory Position and in Appendix A to this guide. Additional information relative to the above concerns may be found in Revision 1 of NUREG-0897, NUREG/CR-2758, NUREG/CR-2759, NUREG/CR-2760, NUREG/CR-2761, NUREG/CR-2772, NUREG/CR-2791, NUREG/CR-2792, NUREG/CR-2982, NUREG/CR-3170, NUREG/CR-3394, NUREG/CR-3616, NUREG/CR-6224, NUREG/CR-6369, NUREG/CR-6762, NUREG/CR-6772, NUREG/CR-6773, NRC Information Notice 94-57, NRC Information Notice 95-06, NRC Information Notice 95-47, Regulatory Guide 1.1, Safety Evaluation on the BWROG's URG for ECCS Suction Strainer Blockage, NEDO-32686, Generic Letter 97-04, and Generic Letter 98-04. A current knowledge base describing results of research on the BWR suction-strainer and PWR sump screen blockage is provided in NUREG/CR-6808.

This regulatory guide provides separate guidance for PWR and BWR plants based on the design features of currently operating reactors. Advanced PWR or BWR designs may employ

design features that this regulatory guide only associates with the opposite reactor design (e.g., an advanced PWR design that employs an in-containment refueling water storage tank that is similar to the suppression pool of a current BWR design, or an advanced BWR design that employs a large dry containment that is similar to a current PWR design). Therefore, for advanced PWR and BWR designs, the guidance provided in both the PWR and BWR sections of this regulatory guide that is appropriate and consistent with the plant's design features should be considered.

In the process of resolving the strainer blockage issue associated with existing BWRs, advanced strainer designs were developed (e.g., stacked disk strainer) and installed. Similarly, it is anticipated that alternative sump screen designs may be developed for the currently licensed PWRs. Much of the guidance in this regulatory guide relates to sump designs currently in use by PWR plants (e.g., vertically or horizontally oriented screens). However, the regulatory guide is not intended to show any preference toward a particular sump screen orientation. The concepts and the intent of the guidance apply to all alternative sump screen designs.

PRESSURIZED WATER REACTORS

In PWRs, the containment emergency sumps provide for the collection of reactor coolant and chemically reactive spray solutions following a LOCA; thus, the sumps serve as water sources to support long-term recirculation for the functions of residual heat removal, emergency core cooling, containment cooling, and containment atmosphere cleanup. These water sources, the related pump inlets, and the piping between the sources and inlets are important safety components. In this guide, the term ECCS implicitly includes the containment spray systems (CSS), and the sumps servicing the ECCS and the CSS are referred to as ECC sumps. Features and relationships of the ECC sumps pertinent to this guide are shown in Figure 1. In operating PWRs, the ECC sump designs may vary from this figure (e.g., in some plants sump screens may be located below the floor level). A more comprehensive description of various ECC sump designs is included in NUREG/CR-6762.

The design of PWR sumps and their outlets includes consideration of the avoidance of air ingestion and other undesirable hydraulic effects (e.g., circulatory flow patterns, outlets leading to high head losses). The location and size of the sump outlets within ECC sumps is important in order to minimize air ingestion since ingestion is a function of submergence level and velocity in the outlet piping. It has been experimentally determined for PWRs that air ingestion can be minimized or eliminated if the sump hydraulic design considerations provided in Appendix A to this guide are followed. Revision 1 of NUREG-0897, NUREG/CR-2758, NUREG/CR-2761, and NUREG/CR-2792 provide additional technical information relevant to sump ECC hydraulic performance and design guidelines.

In order for a centrifugal pump to perform its safety function, there must be adequate margin between the available NPSH and the required NPSH.[1] The available NPSH is the total suction head of liquid absolute, determined at the first stage impeller datum, less the absolute

[1]ANSI/HI 1.1-1.5-1994, "American National Standard for Centrifugal Pumps for Nomenclature, Definitions, Application and Operation," which this guide references, defines NPSH parameters, including required NPSH. See Appendix A of this guide. Required NPSH, as defined in in ANSI/HI 1.1-1.5-1994, is not an NRC regulatory requirement.

vapor pressure of the liquid. The required NPSH is the amount of suction head, over vapor pressure, required to prevent more than 3% loss in total head of the first stage of the pump at a specific capacity.

NPSH margin is the amount by which available NPSH exceeds required NPSH. Failure to provide and maintain adequate NPSH margin for the ECCS pumps could result in cavitation and their subsequent failure to deliver the amount of water assumed in design basis LOCA calculations. Failure to provide and maintain adequate NPSH margin for the containment heat removal pumps could result in pressurization of the containment above the design pressure and an increase in the offsite and control room radiological doses.

The head loss due to debris is not included in the definition of the available NPSH. The value of the head loss due to debris should be compared to the value of the NPSH margin in order to determine whether pump cavitation will occur.

Predicted performance of the ECCS and the containment heat removal pumps should be independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions. For example, if proper operation of the ECCS or the containment heat removal system depends on containment pressure above a specified minimum amount, operation of these systems at a containment pressure less than this amount (resulting, for example, from impaired containment integrity or operation of the containment heat removal systems at too high a rate) could significantly affect the ability of this system to accomplish its safety functions. However, for some operating reactors, some credit for containment accident pressure may be necessary. This should be minimized to the extent possible.

ANSI/HI 1.1-1.5-1994 specifies a method of accounting for the decrease in required NPSH with an increase in temperature of the pumped fluid. This method is subject to restrictions specified in the standard dealing with experience with the specific pump, the amount of air dissolved in the fluid, and the transient nature of the pressure and temperature of the pumped fluid. The staff considers it prudent to not take credit for the reduction in required NPSH that is due to the temperature of the pumped fluid because of the uncertainty in these factors.

Transient NPSH calculations should be performed to ensure that the most limiting conditions are chosen and that the results are conservative.

Placement of the ECC sumps at the lowest level practical ensures maximum use of available recirculation coolant. Areas within the containment in which coolant could accumulate during the containment spray period are provided, as necessary, with drains or flow paths to the sumps to prevent coolant holdup. It is also a concern that these drains or flow paths may themselves be blocked either totally or partially, diverting water away from the active sump region. This guide does not address the design of such drains or flow paths. Because debris can migrate to the sump via these drains or paths, they are best terminated in a manner that will prevent debris from being transported to and accumulating on or within the ECC sumps.

Containment drainage sumps are used to collect and monitor normal leakage flow for leakage detection systems within containments. They are separated from the ECC sumps and are located at an elevation lower than the ECC sumps to minimize inadvertent spillover into the ECC sumps from minor leaks or spills within containment. The floor adjacent to the ECC sumps would normally slope downward, away from the ECC sumps, toward the drainage collection sumps. This downward slope away from the ECC sumps will minimize the transport and collection of debris against the debris interceptors. High-density debris may be swept along the floor by the flow toward the trash rack. A debris curb upstream of and in close proximity to the rack will decrease the amount of such debris reaching the trash rack and debris screens. Debris blockage of the sump screen may also be mitigated by placement of an active device or system that performs an active function to prevent debris (which could block restrictions or damage components in the systems served by the ECC pumps) from entering the ECC pump suction lines, to remove debris from the sump screen and flow stream upstream of the ECC pumps, or to mitigate any detrimental effects of debris accumulation. Examples of active mitigation systems are listed in Appendix B.

It is necessary to protect sump outlets with sump screens and trash racks of sufficient strength to withstand the vibratory motion of seismic events, to resist jet loads and impact loads that could be imposed by missiles that may be generated by the initial LOCA, and to withstand the differential pressure loads imposed by the accumulation of debris. Considerations for selecting materials for the debris interceptors include long periods of inactivity, i.e., no submergence, and periods of operation involving partial or full submergence in a fluid that may contain chemically reactive materials. Isolation of the ECC sumps from high-energy pipe lines is an important consideration in protection against missiles, and it is necessary to shield the screens and racks adequately from impacts of ruptured high-energy piping and associated jet loads. When the screen and rack structures are oriented vertically or nearly vertically, the adverse effects from large pieces of debris (e.g., partially torn insulation blankets or damaged reflective metallic insulation cassettes) collecting on them will be reduced. Consistent with the plant licensing basis single-failure criterion, redundant ECC sumps and sump outlets should be separated to the extent practical to reduce the possibility that a single event could render both sumps inoperable.

It is generally expected that the water surface will be above the top of the debris interceptor structure after completion of the safety injection and before the ECC sumps become operational. However, the uncertainties about the extent of water coverage on the structure, the amount of floating debris that may accumulate, and the potential for early clogging do not favor the use of a horizontal top interceptor. Therefore, in the computation of available interceptor surface area, no credit may be taken for any horizontal interceptor surface unless plant evaluations that adequately account for inherent water source uncertainties demonstrate that the horizontal surface will be submerged at the time of recirculation. For certain sump designs, it is preferable that the top of the interceptor structure be a solid cover plate that will provide additional protection from LOCA-generated loads and be designed to provide for the venting of any trapped air. It is possible that ECC sumps in some plants may not be submerged completely under water at the time of recirculation, either because of unique sump designs or uncertainties in water level estimates. Such partially submerged sumps may be subject to failure criteria other than NPSH margin as discussed in Regulatory Position 1.3.4.4 and Appendix A of this guide. In the case of partially

submerged sumps, credit should only be given for the portion of the sump screen that is expected to be submerged as a function of time.

All debris that is transportable to the trash rack, the debris screen, and the outlets needs to be analyzed for head loss effects. Debris that is small enough to pass through the trash rack and the debris screen needs to be analyzed for head loss effects together with the fibrous debris bed that may filter small particulates. Blockage of the trash rack, sump screen, and sump outlet is a function of the types, combinations, sizes, shapes, and quantities of insulation debris that can be transported to these components. A vertical or nearly vertical inner debris screen located above the containment floor level would minimize the deposition or settling of debris on screen surfaces and thus help to ensure the greatest possible free flow through the fine inner debris screen. Similarly, locating the sump screens and trash racks above the containment floor level, preferably on a pedestal, minimizes the potential for debris buildup. NUREG/CR-6773 provides test results for transport of various types, sizes, and shapes of debris.

The size of openings in the screens is dependent on the physical restrictions that may exist in the systems that are supplied with coolant from the ECC sump. The size of the mesh of the fine debris screen is determined by considering a number of factors, including the size of the openings in the containment spray nozzles; coolant channel openings in the core fuel assemblies; the presence of fuel assembly inlet debris screens; the minimum dimension within the flow-path (e.g., high pressure safety injection (HPSI) throttle valves); such pump design characteristics as seals, bearings, and impeller running clearances; the clean screen head loss; and the consequences of the downstream accumulation of debris passing through the sump screen.

As noted above, degraded pumping can be caused by a number of factors, including plant design and layout. In particular, debris blockage effects on debris interceptor and sump outlet configurations and post-LOCA hydraulic conditions (e.g., air ingestion) must be considered in a combined manner. Small amounts of air ingestion, i.e., 2% or less, will not lead to severe pumping degradation if the required NPSH from the pump manufacturer's curves is increased based on the calculated air ingestion. Thus it is important to use the combined results of all post-LOCA effects to estimate NPSH margin as calculated for the pump inlet. Appendix A to this guide provides information for estimating NPSH margins in PWR sump designs where estimated levels of air ingestion are low (2% or less). Revision 1 of NUREG-0897 and NUREG/CR-2792 provide additional technical findings relevant to NPSH effects on pumps performing the functions of residual heat removal, emergency core cooling, and containment atmosphere cleanup. When air ingestion is 2% or less, compensation for its effects may be achieved without redesign if the available NPSH is greater than the required NPSH plus a margin based on the percentage of air ingestion. If air ingestion is not small, redesign of one or more of the recirculation loop components may be necessary.

BOILING WATER REACTORS

In BWRs, the suppression pool, in conjunction with the primary containment, downcomers, and vents, serves as the water source for effecting long-term recirculation cooling. This source, the related pump suction inlets, and the piping between them are important safety

components. Features and relationships of the suppression pool pertinent to this guide are shown in Figure 2. Concerns with the performance of the suppression pool hydraulics and ECC pump suction strainers include consideration of air ingestion effects, blockage of suction strainers (by debris), and the combined effects of these items on the operability of the ECC pumps (e.g., the impact on NPSH available at the pump inlets). Revision 1 of NUREG-0897 and NUREG/CR-2772 provide data on the performance and air ingestion characteristics of BWR suction strainer configurations.

In order for a centrifugal pump to perform its safety function, there must be adequate margin between the available and the required NPSH. Failure to provide and maintain adequate NPSH for the emergency core cooling system pumps could result in cavitation and their subsequent failure to deliver the amount of water assumed in design basis LOCA calculations. For BWRs that credit containment spray systems in the safety analyses, failure to provide and maintain adequate NPSH of the containment heat removal pumps could result in overpressurization of the containment and an increase in the offsite and control room radiological dose.

Since the safety of a nuclear power plant depends on the expected performance of the centrifugal pumps in the ECCS and the containment heat removal system, it is important to maintain adequate margin between the available and required NPSH under all potential conditions.

The available NPSH is the total suction head of liquid absolute, determined at the first stage impeller datum, less the absolute vapor pressure of the liquid. The required NPSH is the amount of suction head, over vapor pressure, required to prevent more than 3% loss in total head of the first stage of the pump at a specific capacity.

Predicted performance of the ECC and the containment heat removal pumps should be independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions. For example, if proper operation of the ECCS or the containment heat removal system depends on containment pressure above a specified minimum amount, operation of these systems at a containment pressure less than this amount (resulting, for example, from impaired containment integrity or operation of the containment heat removal systems at too high a rate) could significantly affect the ability of this system to accomplish its safety functions. However, for some operating reactors, credit for containment accident pressure may be necessary. This should be minimized to the extent possible.

ANSI/HI 1.1-1.5-1994 specifies a method of accounting for the decrease in required NPSH with an increase in temperature of the pumped fluid. This method is subject to restrictions specified in the standard dealing with experience with the specific pump, the amount of air dissolved in the fluid, and the transient nature of the pressure and temperature of the pumped fluid. The staff has considered it prudent to not take credit for the reduction in required NPSH that is due to the temperature of the pumped fluid because of the uncertainty in these factors.

Transient NPSH calculations should be performed to ensure that the most limiting conditions are chosen and that the results are conservative.

It is desirable to consider the use of debris interceptors (i.e., suction strainers) in BWR designs to protect the pump inlets and NPSH margins. The debris interceptor can be a passive suction strainer or an active suction strainer or active strainer system. A passive suction strainer is a device that prevents debris, which may block restrictions in the systems served by the ECC pumps or damage components, from entering the ECC pump suction line by accumulating debris on a porous surface. An example of a passive suction strainer is a truncated-cone-shaped, perforated plate strainer. An active suction strainer or an active strainer system is a device or system that will take some action to prevent debris, which may block restrictions in the systems served by the ECC pumps or damage components, from entering the ECC pump suction lines, remove debris from the flow stream upstream of the ECC pumps, or mitigate any detrimental effects of debris accumulation. Examples of active mitigation systems are listed in Appendix B.

Suppression pool debris transport analysis should include the effects of LOCA progression because LOCAs of different sizes will affect the duration of LOCA-related hydrodynamic phenomena (e.g., condensation oscillation, chugging). The LOCA-related hydrodynamic phenomena and long-term recirculation hydrodynamic conditions will affect the transport of debris in the suppression pool.

Debris that is transported to the suppression pool during a LOCA, or that is present in the suppression pool prior to a LOCA (NRC Information Notices 94-57, 95-06, and 95-47), could block or damage the suction strainers and needs to be analyzed for head loss effects. This head loss analysis should include filtering of particulate debris by the accumulated debris bed. The head loss characteristics of a debris bed will be a function of the types and quantities of the debris, suction strainer approach velocities, and LOCA-related hydrodynamic phenomena in the suppression pool.

C. REGULATORY POSITION

This section states regulatory positions on design criteria, performance standards, and analysis methods that relate to PWRs (Regulatory Position 1) and BWRs (Regulatory Position 2). As stated in the Introduction to this guide, the purpose of the guidance is to identify information and methods acceptable to the NRC staff for evaluating analytical techniques and implementing regulations related to water sources for long-term cooling of both existing and future reactor systems. The guidance, to a great extent, is generic and it may go beyond the current design of some operating reactor systems.

1. PRESSURIZED WATER REACTORS

1.1 Features Needed To Minimize the Potential for Loss of NPSH

The ECC sumps, which are the source of water for such functions as ECC and containment heat removal following a LOCA, should contain an appropriate combination of the following features and capabilities to ensure the availability of the ECC sumps for long-term cooling. The

adequacy of the combinations of the features and capabilities should be evaluated using the criteria and assumptions in Regulatory Position 1.3.

1.1.1 ECC Sumps, Debris Interceptors, and Debris Screens

1.1.1.1 A minimum of two sumps should be provided, each with sufficient capacity to service one of the redundant trains of the ECCS and CSS. Distribution of water sources and containment spray between the sumps should be considered in the calculation of boron concentration in the sumps for evaluating post-LOCA subcriticality and shutdown margins. Typically, these calculations are performed assuming minimum boron concentration and minimum dilution sources. Similar considerations should also be given in the calculation of time for Hot Leg Switchover, which is calculated assuming maximum boron concentration and a minimum of dilution sources.

1.1.1.2 To the extent practical, the redundant sumps should be physically separated by structural barriers from each other and from high-energy piping systems to preclude damage from LOCA, and, if within the design basis, main steam or main feedwater break consequences to the components of both sumps (e.g., trash racks, sump screens, and sump outlets) by whipping pipes or high-velocity jets of water or steam.

1.1.1.3 The sumps should be located on the lowest floor elevation in the containment exclusive of the reactor vessel cavity to maximize the pool depth relative to the sump screens. The sump outlets should be protected by appropriately oriented (e.g., at least two vertical or nearly vertical) debris interceptors: (1) a fine inner debris screen and (2) a coarse outer trash rack to prevent large debris from reaching the debris screen. A curb should be provided upstream of the trash racks to prevent high-density debris from being swept along the floor into the sump. To be effective, the height of the curb should be appropriate for the pool flow velocities, as the debris can jump over a curb if the velocities are sufficiently high. Experiments documented in NUREG/CR-6772 and NUREG/CR-6773 have demonstrated that substantial quantities of settled debris could transport across the sump pool floor to the sump screen by sliding or tumbling.

1.1.1.4 The floor in the vicinity of the ECC sump should slope gradually downward away from the sump to further retard floor debris transport and reduce the fraction of debris that might reach the sump screen.

1.1.1.5 All drains from the upper regions of the containment should terminate in such a manner that direct streams of water, which may contain entrained debris, will not directly impinge on the debris interceptors or discharge in close proximity to the sump. The drains and other narrow pathways that connect compartments with potential break locations to the ECC sump should be designed to ensure that they would not become blocked by the debris; this is to ensure that water needed for an adequate NPSH margin could not be held up or diverted from the sump.

1.1.1.6 The strength of the trash racks should be adequate to protect the debris screens from missiles and other large debris. Trash racks and sump screens should be capable of withstanding the loads imposed by expanding jets, missiles, the accumulation of debris, and pressure differentials caused by post-LOCA blockage under design-basis flow conditions. When evaluating impact from potential expanding jets and missiles, credit for any protection to trash racks and sump screens offered by surrounding structures or credit for remoteness of trash racks and sump screens from potential high energy sources should be justified.

1.1.1.7 Where consistent with overall sump design and functionality, the top of the debris interceptor structures should be a solid cover plate that is designed to be fully submerged after a LOCA and completion of the ECC injection. The cover plate is intended to provide additional protection to debris interceptor structures from LOCA-generated loads. However, the design should also provide means for venting of any air trapped underneath the cover.

1.1.1.8 The debris interceptors should be designed to withstand the inertial and hydrodynamic effects that are due to vibratory motion of a safe shutdown earthquake (SSE) following a LOCA without loss of structural integrity.

1.1.1.9 Materials for debris interceptors and sump screens should be selected to avoid degradation during periods of both inactivity and operation and should have a low sensitivity to such adverse effects as stress-assisted corrosion that may be induced by chemically reactive spray during LOCA conditions.

1.1.1.10 The debris interceptor structures should include access openings to facilitate inspection of these structures, any vortex suppressors, and the sump outlets.

1.1.1.11 A sump screen design (i.e., size and shape) should be chosen that will avoid the loss of NPSH from debris blockage during the period that the ECCS is required to operate in order to maintain long-term cooling or maximize the time before loss of NPSH caused by debris blockage when used with an active mitigation system (see Regulatory Position 1.1.4).

1.1.1.12 The possibility of debris-clogging flow restrictions downstream of the sump screen should be assessed to ensure adequate long term recirculation cooling, containment cooling, and containment pressure control capabilities. The size of the openings in the sump debris screen should be determined considering the flow restrictions of systems served by the ECCS sump. The potential for long thin slivers passing axially through the sump screen and then reorienting and clogging at any flow restriction downstream should be considered.

Consideration should be given to the buildup of debris at downstream locations such as the following: containment spray nozzle openings, HPSI throttle valves, coolant channel openings in the core fuel assemblies, fuel assembly inlet debris screens, ECCS

pump seals, bearings, and impeller running clearances. If it is determined that a sump screen with openings small enough to filter out particles of debris that are fine enough to cause damage to ECCS pump seals or bearings would be impractical, it is expected that modifications would be made to ECCS pumps or ECCS pumps would be procured that can operate long term under the probable conditions.

1.1.1.13 ECC and containment spray pump suction inlets should be designed to prevent degradation of pump performance through air ingestion and other adverse hydraulic effects (e.g., circulatory flow patterns, high intake head losses).

1.1.1.14 All drains from the upper regions of the containment building, as well as floor drains, should terminate in such a manner that direct streams of water, which may contain entrained debris, will not discharge downstream of the sump screen, thereby bypassing the sump screen.

1.1.1.15 Advanced strainer designs (e.g., stacked disc strainers) have demonstrated capabilities that are not provided by simple flat plate or cone-shaped strainers or screens. For example, these capabilities include built-in debris traps where debris can collect on surfaces while keeping a portion of the screen relatively free of debris. The convoluted structure of such strainer designs increases the total screen area, and these structures tend to prevent the condition referred to as the thin bed effect. It may be desirable to include these capabilities in any new sump strainer/screen designs. The performance characteristics and effectiveness of such designs should be supported by appropriate test data for any particular intended application.

1.1.2 Minimizing Debris

The debris (see Regulatory Position 1.3.2) that could accumulate on the sump screen should be minimized.

1.1.2.1 Cleanliness programs should be established to clean the containment on a regular basis, and plant procedures should be established for control and removal of foreign materials from the containment.

1.1.2.2 Insulation types (e.g., fibrous and calcium silicate) that can be sources of debris that is known to more readily transport to the sump screen and cause higher head losses may be replaced with insulations (e.g., reflective metallic insulation) that transport less readily and cause less severe head losses once deposited onto the sump screen. If insulation is replaced or otherwise removed during maintenance, abatement procedures should be established to avoid generating latent debris in the containment.

1.1.2.3 To minimize potential debris caused by chemical reaction of the pool water with metals in the containment, exposure of bare metal surfaces (e.g., scaffolding) to containment cooling water through spray impingement or immersion should be minimized either by removal or by chemical-resistant protection (e.g., coatings or jackets).

1.1.3 Instrumentation

If relying on operator actions to mitigate the consequences of the accumulation of debris on the ECC sump screens, safety-related instrumentation that provides operators with an indication and audible warning of impending loss of NPSH for ECCS pumps should be available in the control room.

1.1.4 Active Sump Screen System

An active device or system (see examples in Appendix B) may be provided to prevent the accumulation of debris on a sump screen or to mitigate the consequences of accumulation of debris on a sump screen. An active system should be able to prevent debris that may block restrictions found in the systems served by the ECC pumps from entering the system. The operation of the active component or system should not adversely affect the operation of other ECC components or systems. Performance characteristics of an active sump screen system should be supported by appropriate test data that address head loss performance.

1.1.5 Inservice Inspection

To ensure the operability and structural integrity of the trash racks and screens, access openings are necessary to permit inspection of the ECC sump structures and outlets. Inservice inspection of racks, screens, vortex suppressors, and sump outlets, including visual examination for evidence of structural degradation or corrosion, should be performed on a regular basis at every refueling period downtime. Inspection of the ECC sump components late in the refueling period will ensure the absence of construction trash in the ECC sump area.

1.2 Evaluation of Alternative Water Sources

To demonstrate that a combination of the features and actions listed above are adequate to ensure long-term cooling and that the five criteria of 10 CFR 50.46(b) will be met following a LOCA, an evaluation using the guidance and assumptions in Regulatory Position 1.3 should be conducted. If a licensee is relying on operator actions to prevent the accumulation of debris on ECC sump screens or to mitigate the consequences of the accumulation of debris on the ECC sump screens, an evaluation should be performed to ensure that the operator has adequate indications, training, time, and system capabilities to perform the necessary actions. If not covered by plant-specific emergency operating procedures, procedures should be established to use alternative water sources that will be activated when unacceptable head loss renders the sump inoperable. The valves needed to align the ECCS and containment spray systems (taking suction from the recirculation sumps) with an alternative water source should be periodically inspected and maintained.

1.3 Evaluation of Long-Term Recirculation Capability

The following techniques, assumptions, and guidance should be used in a deterministic, plant-specific evaluation to ensure that any implementation of a combination of the features and capabilities listed in Regulatory Position 1.1 are adequate to ensure the availability of a reliable water source for long-term recirculation following a LOCA. The assumptions and guidance listed below can also be used to develop test conditions for sump screens.

Evaluation and confirmation of (1) sump hydraulic performance (e.g., geometric effects, air ingestion), (2) debris effects (e.g., debris transport, interceptor blockage, head loss), and (3) the combined impact on NPSH available at the pump inlet should be performed to ensure that long-term recirculation cooling can be accomplished following a LOCA. Such an evaluation should arrive at a determination of NPSH margin calculated at the pump inlet. An assessment should also be made of the susceptibility to debris blockage of the containment drainage flow paths to the recirculation sump; this is to protect against reduction in available NPSH if substantial amounts of water are held up or diverted away from the sump. An assessment should be made of the susceptibility of the flow restrictions in the ECCS and CSS recirculation flow paths downstream of the sump screens and of the recirculation pump seal and bearing assembly design to failure from particulate ingestion and abrasive effects to protect against degradation of long-term recirculation pumping capacity.

1.3.1 Net Positive Suction Head of ECCS and Containment Heat Removal Pumps

1.3.1.1 ECC and containment heat removal systems should be designed so that sufficient available NPSH is provided to the system pumps, assuming the maximum expected temperature of pumped fluid and no increase in containment pressure from that present prior to the postulated LOCA. (See Regulatory Position 1.3.1.2.)

For sump pools with temperatures less than 212 °F, it is conservative to assume that the containment pressure equals the vapor pressure of the sump water. This ensures that credit is not taken for the containment pressurization during the transient.

For subatmospheric containments, this guidance should apply after the injection phase has terminated. For subatmospheric containments, prior to termination of the injection phase, NPSH analyses should include conservative predictions of the containment atmospheric pressure and sump water temperature as a function of time.

1.3.1.2 For certain operating PWRs for which the design cannot be practicably altered, conformance with Regulatory Position 1.3.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. Calculation of available containment pressure and sump water temperature as a function of time should underestimate the expected containment pressure and overestimate the sump water temperature when determining available NPSH for this situation.

1.3.1.3 For certain operating reactors for which the design cannot be practicably altered, if credit is taken for operation of an ECCS or containment heat removal pump in cavitation, prototypical pump tests should be performed along with post-test examination of the pump to demonstrate that pump performance will not be degraded and that the pump continues to meet all the performance criteria assumed in the safety analyses. The time period in the safety analyses during which the pump may be assumed to operate while cavitating should not be longer than the time for which the performance tests demonstrate that the pump meets performance criteria.

1.3.1.4 The decay and residual heat produced following accident initiation should be included in the determination of the water temperature. The uncertainty in the determination of the decay heat should be included in this calculation. The residual heat should be calculated with margin.

1.3.1.5 The hot channel correction factor specified in ANSI/HI 1.1-1.5-1994 should not be used in determining the margin between the available and required NPSH for ECCS and containment heat removal system pumps.

1.3.1.6 The calculation of available NPSH should minimize the height of water above the pump suction (i.e., the level of water on the containment floor). The calculated height of water on the containment floor should not consider quantities of water that do not contribute to the sump pool (e.g., atmospheric steam, pooled water on floors and in refueling canals, spray droplets and other falling water, etc.). The amount of water in enclosed areas that cannot be readily returned to the sump should not be included in the calculated height of water on the containment floor.

1.3.1.7 The calculation of pipe and fitting resistance and the calculation of the nominal screen resistance without blockage by debris should be done in a recognized, defensible method or determined from applicable experimental data.

1.3.1.8 Sump screen flow resistance that is due to blockage by LOCA-generated debris or foreign material in the containment which is transported to the suction intake screens should be determined using Regulatory Position 1.3.4.

1.3.1.9 Calculation of available NPSH should be performed as a function of time until it is clear that the available NPSH will not decrease further.

1.3.2 Debris Sources and Generation

1.3.2.1 Consistent with the requirements of 10 CFR 50.46, debris generation should be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. The level of severity corresponding to each postulated break should be based on the potential head loss incurred across the sump screen. Some PWRs may need recirculation from the sump for licensing basis events other than LOCAs. Therefore, licensees should evaluate the licensing basis and include potential break locations in the main steam and main feedwater lines as well in determining the most limiting conditions for sump operation.

1.3.2.2 An acceptable method for estimating the amount of debris generated by a postulated LOCA is to use the zone of influence (ZOI). Examples of this approach are provided in NUREG/CR-6224 and Boiling Water Reactor Owners' Group (BWROG) Utility Resolution Guidance (NEDO-32686 and the staff's Safety Evaluation on the BWROG's

response to NRC Bulletin 96-03). A representation of the ZOI for commonly used insulation materials is shown in Figure 3.

- The size and shape of the ZOI should be supported by analysis or experiments for the break and potential debris. The size and shape of the ZOI should be consistent with the debris source (e.g., insulation, fire barrier materials, etc.) damage pressures, i.e., the ZOI should extend until the jet pressures decrease below the experimentally determined damage pressures appropriate for the debris source.

- The volume of debris contained within the ZOI should be used to estimate the amount of debris generated by a postulated break.

- The size distribution of debris created in the ZOI should be determined by analysis or experiments.

- The shock wave generated during the postulated pipe break and the subsequent jet should be the basis for estimating the amount of debris generated and the size or size distribution of the debris generated within the ZOI.

Certain types of material used in a small quantity inside the containment can, with adequate justification, be demonstrated to make a marginal contribution to the debris loading for the ECC sump. If debris generation and debris transport data have not been determined experimentally for such material, it may be grouped with another like material existing in large quantities. For example, a small quantity of fibrous filtering material may be grouped with a substantially large quantity of fibrous insulation debris, and the debris generation and transport data for the filter material need not be determined experimentally. However, such analyses are valid only if the small quantity of material treated in this manner does not have a significant effect when combined with other materials (e.g., a small quantity of calcium silicate combined with fibrous debris).

1.3.2.3 A sufficient number of breaks in each high-pressure system that relies on recirculation should be considered to reasonably bound variations in debris generation by the size, quantity, and type of debris. As a minimum, the following postulated break locations should be considered.

- Breaks in the reactor coolant system (e.g., hot leg, cold leg, pressurizer surge line) and, depending on the plant licensing basis, main steam and main feedwater lines with the largest amount of potential debris within the postulated ZOI,
- Large breaks with two or more different types of debris, including the breaks with the most variety of debris, within the expected ZOI,
- Breaks in areas with the most direct path to the sump,
- Medium and large breaks with the largest potential particulate debris to insulation ratio by weight, and

- Breaks that generate an amount of fibrous debris that, after its transport to the sump screen, could form a uniform thin bed that could subsequently filter sufficient particulate debris to create a relatively high head loss referred to as the 'thin-bed effect.' The minimum thickness of fibrous debris needed to form a thin bed has typically been estimated at 1/8 inch thick based on the nominal insulation density (NUREG/CR-6224).

1.3.2.4 All insulation (e.g., fibrous, calcium silicate, reflective metallic), painted surfaces, fire barrier materials, and fibrous, cloth, plastic, or particulate materials within the ZOI should be considered a debris source. Analytical models or experiments should be used to predict the size of the postulated debris. For breaks postulated in the vicinity of the pressure vessel, the potential for debris generation from the packing materials commonly used in the penetrations and the insulation installed on the pressure vessel should be considered. Particulate debris generated by pipe rupture jets stripping off paint or coatings and eroding concrete at the point of impact should also be considered.

1.3.2.5 The cleanliness of the containment during plant operation should be considered when estimating the amount and type of debris available to block the ECC sump screens. The potential for such material (e.g., thermal insulation other than piping insulation, ropes, fire hoses, wire ties, tape, ventilation system filters, permanent tags or stickers on plant equipment, rust flakes from unpainted steel surfaces, corrosion products, dust and dirt, latent individual fibers) to impact head loss across the ECC sump screens should also be considered.

1.3.2.6 In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) should be considered in the analyses. Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

1.3.2.7 Debris generation that is due to continued degradation of insulation and other debris when subjected to turbulence caused by cascading water flows from upper regions of the containments or near the break overflow region should be considered in the analyses.

1.3.3 Debris Transport

1.3.3.1 The calculation of debris quantities transported from debris sources to the sump screen should consider all modes of debris transport, including airborne debris transport, containment spray washdown debris transport, and containment sump pool debris transport. Consideration of the containment pool debris transport should include (1) debris transport during the fill-up phase, as well as during the recirculation phase, (2) the turbulence in the pool caused by the flow of water, water entering the pool from break overflow, and containment spray drainage, and (3) the buoyancy of the debris. Transport analyses of debris should consider: (1) debris that would float along the pool

surface, (2) debris that would remain suspended due to pool turbulence (e.g., individual fibers and fine particulates), and (3) debris that readily settles to the pool floor.

1.3.3.2 The debris transport analyses should consider each type of insulation (e.g., fibrous, calcium silicate, reflective metallic) and debris size (e.g., particulates, fibrous fine, large pieces of fibrous insulation). The analyses should also consider the potential for further decomposition of the debris as it is transported to the sump screen.

1.3.3.3 Bulk flow velocity from recirculation operations, LOCA-related hydrodynamic phenomena, and other hydrodynamic forces (e.g., local turbulence effects or pool mixing) should be considered for both debris transport and ECC sump screen velocity computations.

1.3.3.4 An acceptable analytical approach to predict debris transport within the sump pool is to use computational fluid dynamics (CFD) simulations in combination with the experimental debris transport data. Examples of this approach are provided in NUREG/CR-6772 and NUREG/CR-6773. Alternative methods for debris transport analyses are also acceptable, provided they are supported by adequate validation of analytical techniques using experimental data to ensure that the debris transport estimates are conservative with respect to the quantities and types of debris transported to the sump screen.

1.3.3.5 Curbs can be credited for removing heavier debris that has been shown analytically or experimentally to travel by sliding along the containment floor and that cannot be lifted off the floor within the calculated water velocity range.

1.3.3.6 If transported to the sump pool, all debris (e.g., fine fibrous, particulates) that would remain suspended due to pool turbulence should be considered to reach the sump screen.

1.3.3.7 The time to switch over to sump recirculation and the operation of containment spray should be considered in the evaluation of debris transport to the sump screen.

1.3.3.8 In lieu of performing airborne and containment spray washdown debris transport analyses, it could be assumed that all debris will be transported to the sump pool.

In lieu of performing sump pool debris transport analyses (Regulatory Position 1.3.3.4), it could be assumed that all debris entering the sump pool or originating in the sump will be considered transported to the sump screen when estimating screen debris bed head loss.

If it is credible in a plant that all drains leading to the containment sump could become completely blocked, or an inventory holdup in containment could happen together with debris loading on the sump screen, these situations could pose a worse impact on the recirculation sump performance than the assumed situations mentioned above. In this case, these situations should also be assessed.

1.3.3.9 The effects of floating or buoyant debris on the integrity of the sump screen and on subsequent head loss should be considered. For screens that are not fully submerged or are only shallowly submerged, floating debris could contribute to the debris bed head loss. The head loss due to floating or buoyant debris could be minimized by a design feature to keep buoyant debris from reaching the sump screen.

1.3.4 Debris Accumulation and Head Loss

1.3.4.1 ECC sump screen blockage should be evaluated based on the amount of debris estimated using the assumptions and criteria described in Regulatory Position 1.3.2 and on the debris transported to the ECC sump per Regulatory Position 1.3.3. This volume of debris should be used to estimate the rate of accumulation of debris on the ECC sump screen.

1.3.4.2 Consideration of ECC sump screen submergence (full or partial) at the time of switchover to ECCS should be given in calculating the available (wetted) screen area. For plants in which containment heat removal pumps take suction from the ECC sump before switchover to the ECCS, the available NPSH for these pumps should consider the submergence of the sump screens at the time these pumps initiate suction from the ECC sump. Unless otherwise shown analytically or experimentally, debris should be assumed to be uniformly distributed over the available sump screen surface. Debris mass should be calculated based on the amount of debris estimated to reach the ECC sump screen. (See Revision 1 of NUREG-0897, NUREG/CR-3616, and NUREG/CR-6224.)

1.3.4.3 For fully submerged sump screens, the NPSH available to the ECC pumps should be determined using the conditions specified in the plant's licensing basis.

1.3.4.4 For partially submerged sumps, NPSH margin may not be the only failure criterion, as discussed in Appendix A. For partially submerged sumps, credit should only be given to the portion of the sump screen that is expected to be submerged, as a function of time. Pump failure should be assumed to occur when the head loss across the sump screen (including only the clean screen head loss and the debris bed head loss) is greater than one-half of the submerged screen height or NPSH margin.

1.3.4.5 Estimates of head loss caused by debris blockage should be developed from empirical data based on the sump screen design (e.g., surface area and geometry), postulated combinations of debris (i.e., amount, size distribution, type), and approach velocity. Because debris beds that form on sump screens can trap debris that would pass through an unobstructed sump screen opening, any head loss correlation should conservatively account for filtration of particulates by the debris bed, including particulates that would pass through an unobstructed sump screen.

1.3.4.6 Consistent with the requirements of 10 CFR 50.46, head loss should be calculated for the debris beds formed of different combinations of fibers and particulate mixtures (e.g.,

minimum uniform thin bed of fibers supporting a layer of particulate debris) based on assumptions and criteria described in Regulatory Positions 1.3.2 and 1.3.3.

2. BOILING WATER REACTORS

2.1 Features Needed To Minimize the Potential for Loss of NPSH

The suppression pool is the source of water for such functions as ECC and containment heat removal following a LOCA, in conjunction with the vents and downcomers between the drywell and the wetwell. It should combine the following features and capabilities to ensure the availability of the suppression pool for long-term cooling. The adequacy of the combinations of the features and capabilities should be evaluated using the criteria and assumptions in Regulatory Position 2.2.

2.1.1 Net Positive Suction Head of ECCS and Containment Heat Removal Pumps

2.1.1.1 ECC and containment heat removal systems should be designed so that adequate available NPSH is provided to the system pumps, assuming the maximum expected temperature of the pumped fluid and no increase in containment pressure from that present prior to the postulated LOCAs. (See Regulatory Position 2.1.1.2.)

2.1.1.2 For certain operating BWRs for which the design cannot be practicably altered, conformance with Regulatory Position 2.1.1.1 may not be possible. In these cases, no additional containment pressure should be included in the determination of available NPSH than is necessary to preclude pump cavitation. Calculation of available containment pressure should underestimate the expected containment pressure when determining available NPSH for this situation. Calculation of suppression pool water temperature should overestimate the expected temperature when determining available NPSH.

2.1.1.3 For certain operating BWRs for which the design cannot be practicably altered, if credit is taken for operation of an ECCS or containment heat removal pump in cavitation, prototypical pump tests should be performed along with post-test examination of the pump to demonstrate that pump performance will not be degraded and that the pump continues to meet all the performance criteria assumed in the safety analyses. The time period in the safety analyses during which the pump may be assumed to operate while cavitating should not be longer than the time for which the performance tests demonstrate the pump meets performance criteria.

2.1.1.4 The decay and residual heat produced following accident initiation should be included in the determination of the water temperature. The uncertainty in the determination of the decay heat should be included in this calculation. The residual heat should be calculated with margin.

2.1.1.5 The hot channel correction factor specified in ANSI/HI 1.1-1.5-1994 should not be used in determining the margin between the available and required NPSH for ECCS and containment heat removal system pumps.

2.1.1.6 The level of water in suppression pools should be the minimum value given in the technical specifications reduced by the drawdown due to suppression pool water in the drywell and the sprays.

2.1.1.7 Pipe and fitting resistance and the nominal screen resistance without blockage by debris should be calculated in a recognized, defensible method or determined from applicable experimental data.

2.1.1.8 Suction strainer screen flow resistance caused by blockage by LOCA-generated debris or foreign material in the containment that is transported to the suction intake screens should be determined using the methods in Regulatory Position 2.3.3.

2.1.1.9 Calculation of available NPSH should be performed as a function of time until it is clear that the available NPSH will not decrease further.

2.1.2 Passive Strainer

The inlet of pumps performing the above functions should be protected by a suction strainer placed upstream of the pumps; this is to prevent the ingestion of debris that may damage components or block restrictions in the systems served by the ECC pumps. The following items should be considered in the design and implementation of a passive strainer.

2.1.2.1 The suction strainer design (i.e., size and shape) should be chosen to avoid the loss of NPSH from debris blockage during the period that the ECCS is required to operate in order to maintain long-term cooling or maximize the time before loss of NPSH caused by debris blockage when used with an active mitigation system (see Regulatory Position 2.1.5).

2.1.2.2 The possibility of debris clogging flow restrictions downstream of the strainers should be assessed to ensure adequate long-term ECCS performance. The size of openings in the suppression pool suction strainers should be based on the minimum restrictions found in systems served by the suppression pool. The potential for long thin slivers passing axially through the strainer and then reorienting and clogging at any flow restriction downstream should be considered.

Consideration should be given to the buildup of debris at the following downstream locations: spray nozzle openings, throttle valves, coolant channel openings in the core fuel assemblies, fuel assembly inlet debris screens, ECCS pump seals, bearings, and impeller running clearances. If it is determined that a strainer with openings small enough to filter out particles of debris that are fine enough to cause damage to ECCS pump seals or bearings would be impractical, it is expected that modifications would be

made to ECCS pumps or ECCS pumps would be procured that can operate long term under the probable conditions.

2.1.2.3 ECC pump suction inlets should be designed to prevent degradation of pump performance through air ingestion and other adverse hydraulic effects (e.g., circulatory flow patterns, high intake head losses).

2.1.2.4 All drains from the upper regions of the containment should terminate in such a manner that direct streams of water, which may contain entrained debris, will not impinge on the suppression pool suction strainers.

2.1.2.5 The strength of the suction strainers should be adequate to protect the debris screen from missiles and other large debris. The strainers and the associated structural supports should be adequate to withstand loads imposed by missiles, debris accumulation, and hydrodynamic loads induced by suppression pool dynamics. To the extent practical, the strainers should be located outside the zone of influence of the vents, downcomers, or spargers to minimize hydrodynamic loads. The strainer design, vis-a-vis the hydrodynamic loads, should be validated analytically or experimentally.

2.1.2.6 The suction strainers should be designed to withstand the inertial and hydrodynamic effects that are due to vibratory motion of a safe shutdown earthquake (SSE) without loss of structural integrity.

2.1.2.7 Material for suction strainers should be selected to avoid degradation during periods of inactivity and operation and should have a low sensitivity to such adverse effects as stress-assisted corrosion that may be induced by coolant during LOCA conditions.

2.1.3 Minimizing Debris

The amount of potential debris (see Regulatory Position 2.3.1) that could clog the ECC suction strainers should be minimized.

2.1.3.1 Containment cleanliness programs should be instituted to clean the suppression pool on a regular basis, and plant procedures should be established for control and removal of foreign materials from the containment.

2.1.3.2 Debris interceptors in the drywell in the vicinity of the downcomers or vents may serve effectively in reducing debris transport to the suppression pool. In addition to meeting Regulatory Position 2.1.2, debris interceptors between the drywell and wetwell should not reduce the suppression capability of the containment.

2.1.3.3 Insulation types (e.g., fibrous and calcium silicate) that can be sources of debris that is known to more readily transport to the strainer and cause higher head losses should be avoided. Insulations (e.g., reflective metallic insulation) that transport less readily and cause less severe head losses once deposited onto the strainers should be used. If

insulation is replaced or otherwise removed during maintenance, abatement procedures should be established to avoid generating latent debris in the containment.

2.1.3.4 To minimize potential debris caused by chemical reaction of coolant with metals in the containment, exposure of bare metal surfaces (e.g., scaffolding) to spray impingement or immersion should be minimized either by removal or by using chemical-resistant protection (e.g., coatings or jackets).

2.1.4 Instrumentation

If relying on operator actions to mitigate the consequences of the accumulation of debris on the suction strainers, safety-related instrumentation that provides operators with an indication and audible warning of impending loss of NPSH for ECCS pumps should be available in the control room.

2.1.5 Active Strainers

An active component or system (see Appendix B) may be provided to prevent the accumulation of debris on a suction strainer or to mitigate the consequences of accumulation of debris on a suction strainer. An active system should be able to prevent debris that may block restrictions found in the systems served by the ECC pumps from entering the system. The operation of the active component or system should not adversely affect the operation of other ECC components or systems. The use of active strainers should be validated by adequate testing.

2.1.6 Inservice Inspection

Inservice inspection requirements should be established that include (1) inspection of the cleanliness of the suppression pool, (2) a visual examination for evidence of structural degradation or corrosion of the suction strainers and strainer system, and (3) an inspection of the wetwell and the drywell, including the vents, downcomers, and deflectors, for the identification and removal of debris or trash that could contribute to the blockage of suppression pool suction strainers. These inservice inspections should be performed on a regular basis at every refueling period downtime.

2.2 Evaluation of Alternative Water Sources

To demonstrate that a combination of the features and actions listed above are adequate to ensure long-term cooling and that the five criteria of 10 CFR 50.46(b) will be met following a LOCA, an evaluation using the guidance and assumptions in Regulatory Position 2.3 should be conducted. If a licensee is relying on operator actions to prevent the accumulation of debris on suction strainers or to mitigate the consequences of the accumulation of debris on the suction strainers, an evaluation should be performed to ensure that the operator has adequate indications, training, time, and system capabilities to perform the necessary actions. If not covered by plant-specific emergency operating procedure, procedures should be established to use alternative water sources. The valves needed to align the ECCS with an alternative water source should be periodically inspected and maintained.

2.3 Evaluation of Long-Term Recirculation Capability

During any evaluation of the susceptibility of a BWR to debris blockage, the considerations and events shown in Figures 4 and 5 should be addressed. The following techniques, assumptions, and guidance should be used in a deterministic evaluation to ensure that any implementation of a combination of the features and capabilities listed in Regulatory Position 2.1 are adequate to ensure the availability of a reliable water source for long-term recirculation after a LOCA. An assessment should be made of the susceptibility to debris blockage of the containment drainage flowpaths to the suppression pool, flow restrictions in the ECCS, and containment spray recirculation flowpaths downstream of the suction strainer to protect against degradation of long-term recirculation pumping capacity. Unless otherwise noted, the techniques, assumptions, and guidance listed below are applicable to an evaluation of passive and active strainers. The assumptions and guidance listed below can also be used to develop test conditions for suction strainers or strainer systems.

2.3.1 Debris Sources and Generation

2.3.1.1 Consistent with the requirements of 10 CFR 50.46, debris generation should be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated.

2.3.1.2 An acceptable method for determining the shape of the zone of influence (ZOI) of a break is described in NUREG/CR-6224 and NEDO-32686. The volume contained within the ZOI should be used to estimate the amount of debris generated by a postulated break. The distance of the ZOI from the break should be supported by analysis or experiments for the break and potential debris. The shock wave generated during postulated pipe break and the subsequent jet should be the basis for estimating the amount of debris generated and the size or size distribution of the debris generated within the ZOI.

Certain types of material used in a small quantity inside the containment can, with adequate justification, be demonstrated to make a marginal contribution to the debris loading for the ECC sump. If debris generation and debris transport data have not been determined experimentally for such material, it may be grouped with another like material existing in large quantities. For example, a small quantity of fibrous filtering material may be grouped with a substantially larger quantity of fibrous insulation debris, and the debris generation and transport data for the filter material need not be determined experimentally. However, such analyses are valid only if the small quantity of material treated in this manner does not have a significant effect when combined with other materials (e.g., a small quantity of calcium silicate combined with fibrous debris).

2.3.1.3 All sources of fibrous materials in the containment such as fire protection materials, thermal insulation, or filters that are present during operation should be identified.

2.3.1.4 All insulation, painted surfaces, and fibrous, cloth, plastic, or particulate materials within the ZOI should be considered debris sources. Analytical models or experiments should be used to predict the size of the postulated debris.

2.3.1.5 A sufficient number of breaks in each high-pressure system that relies on recirculation should be considered to reasonably bound variations in debris generation by the size, quantity, and type of debris. As a minimum, the following postulated break locations should be considered.

- Breaks in the main steam, feedwater, and recirculation lines with the largest amount of potential debris within the postulated ZOI,
- Large breaks with two or more different types of debris, including the breaks with the most variety of debris, within the expected ZOI,
- Breaks in areas with the most direct path between the drywell and wetwell,
- Medium and large breaks with the largest potential particulate debris to insulation ratio by weight, and
- Breaks that generate an amount of fibrous debris that, after its transport to the suction strainer, could form a uniform thin bed that could subsequently filter sufficient particulate debris to create a relatively high head loss referred to as the 'thin-bed effect.' The minimum thickness of fibrous debris needed to form a thin bed has typically been estimated at 1/8 inch thick based on the nominal insulation density (NUREG/CR-6224).

2.3.1.6 The cleanliness of the suppression pool and containment during plant operation should be considered when estimating the amount and type of debris available to block the suction strainers. The potential for such material (e.g., thermal insulation other than piping insulation, ropes, fire hoses, wire ties, tape, ventilation system filters, permanent tags or stickers on plant equipment, rust flakes from unpainted steel surfaces, corrosion products, dust and dirt, latent individual fibers) to impact head loss across the suction strainer should also be considered.

2.3.1.7 The amount of particulates estimated to be in the pool prior to a LOCA should be considered to be the maximum amount of corrosion products (i.e., sludge) expected to be generated since the last time the pool was cleaned. The size distribution and amount of particulates should be based on plant samples.

2.3.1.8 In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) should be considered in the analyses. Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

2.3.2 **Debris Transport**

2.3.2.1 It should be assumed that all debris fragments smaller than the clearances in the gratings will be transported to the suppression pool during blowdown. Credit may be taken for filtration of larger pieces of debris by floor gratings and other interdicting structures present in a drywell (NEDO-32686 and NUREG/CR-6369). However, it should be assumed that a fraction of large fragments captured by the gratings would be eroded by the combined effects of cascading break overflow and the drywell spray flow. The fraction of the smaller debris generated and thus transported to the suppression pool during the blowdown, as well as the fraction of the larger debris that may be eroded during the washdown phase, should be determined analytically or experimentally.

2.3.2.2 It should be assumed that LOCA-induced phenomena (i.e., pool swell, chugging, condensation oscillations) will suspend all the debris assumed to be in the suppression pool at the onset of the LOCA.

2.3.2.3 The concentration of debris in the suppression pool should be calculated based on the amount of debris estimated to reach the suppression pool from the drywell and the amount of debris and foreign materials estimated to be in the suppression pool prior to a postulated break.

2.3.2.4 Credit should not be taken for debris settling until LOCA-induced turbulence in the suppression pool has ceased. The debris settling rate for the postulated debris should be validated analytically or experimentally.

2.3.2.5 Bulk suppression pool velocity from recirculation operations, LOCA-related hydrodynamic phenomena, and other hydrodynamic forces (e.g., local turbulence effects or pool mixing) should be considered for both debris transport and suction strainer velocity computations.

2.3.3 **Strainer Blockage and Head Loss**

2.3.3.1 Strainer blockage should be based on the amount of debris estimated using the assumptions and guidance described in Regulatory Position 2.3.1 and on the debris transported to the wetwell per Regulatory Position 2.3.2. This volume of debris, as well as other materials that could be present in the suppression pool prior to a LOCA, should be used to estimate the rate of accumulation of debris on the strainer surface.

2.3.3.2 The flow rate through the strainer should be used to estimate the rate of accumulation of debris on the strainer surface.

2.3.3.3 The suppression pool suction strainer area used in determining the approach velocity should conservatively account for blockage that may result. Unless otherwise shown analytically or experimentally, debris should be assumed to be uniformly distributed over the available suction strainer surface. Debris mass should be calculated based on

the amount of debris estimated to reach or to be in the suppression pool. (See Revision 1 of NUREG-0897, NUREG/CR-3616, and NUREG/CR-6224.)

2.3.3.4 The NPSH available to the ECC pumps should be determined using the conditions specified in the plant's licensing basis.

2.3.3.5 Estimates of head loss caused by debris blockage should be developed from empirical data based on the strainer design (e.g., surface area and geometry), postulated debris (i.e., amount, size distribution, type), and velocity. Any head loss correlation should conservatively account for filtration of particulates by the debris bed.

2.3.3.6 The performance characteristics of a passive or an active strainer should be supported by appropriate test data that addresses, at a minimum, (1) suppression pool hydrodynamic loads and (2) head loss performance.

D. IMPLEMENTATION

The purpose of this guide is to describe methods acceptable to the NRC staff for analyzing nuclear power plant sumps and suppression pools and for demonstrating their capability to perform long-term recirculation cooling following a loss-of-coolant accident in accordance with the requirements set forth in 10 CFR 50.46.

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[2] Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; telephone (703)487-4650; <http://www.ntis.gov/ordernow>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or (800)397-4209; fax (301)415-3548; email is PDR@NRC.GOV.

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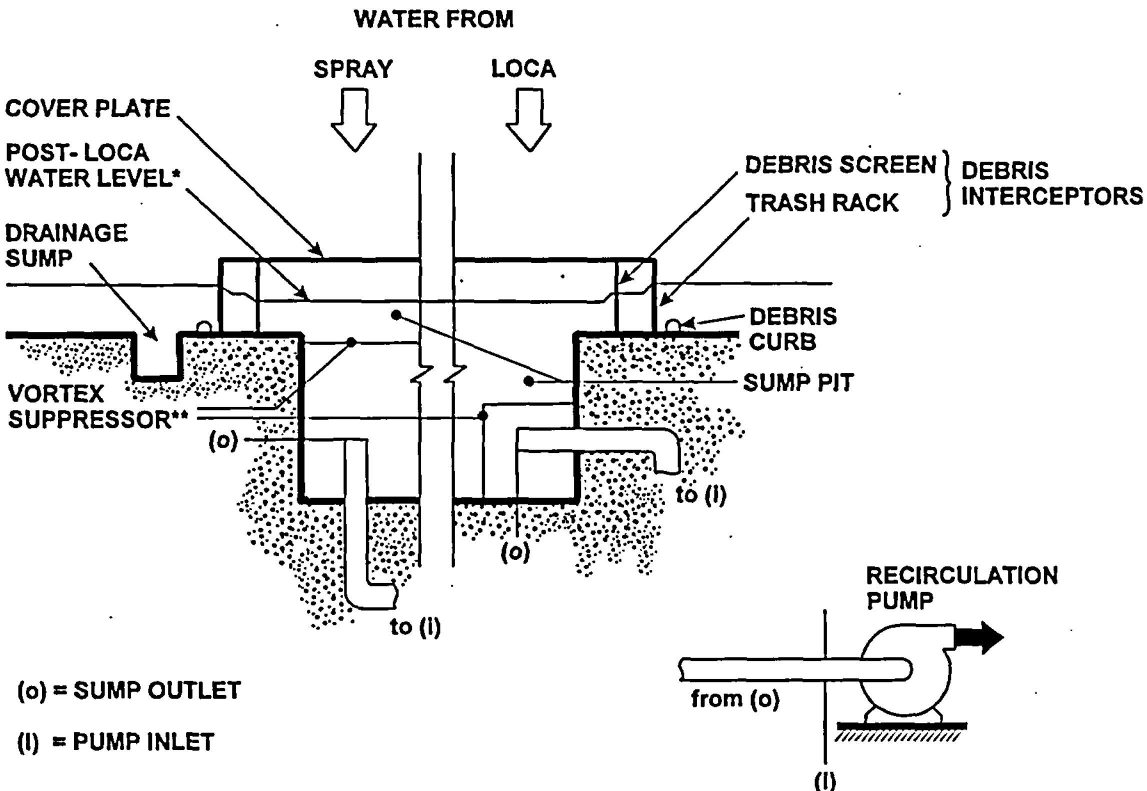
[3] Requests for single copies of draft or active regulatory guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section, or by fax to (301)415-2289; email <DISTRIBUTION@NRC.GOV>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike (first floor), Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or 1-(800)397-4209; fax (301)415-3548; e-mail <PDR@NRC.GOV>.



(o) = SUMP OUTLET

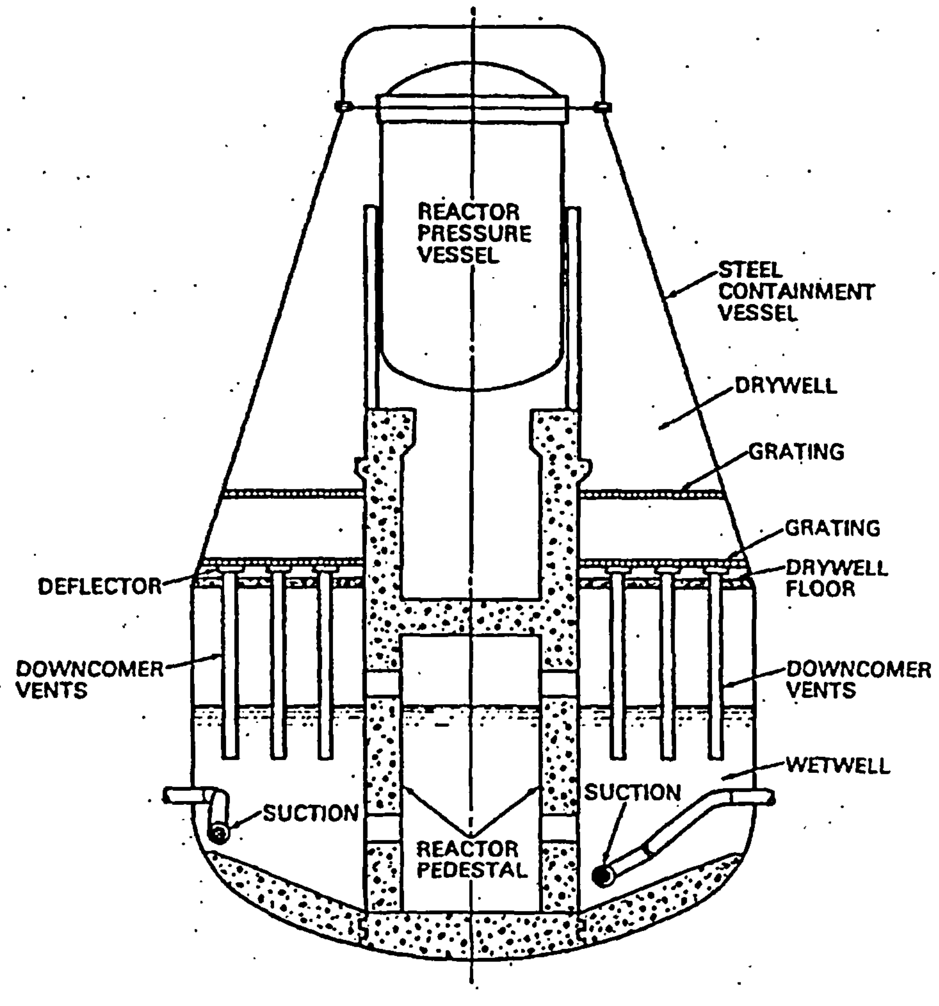
(I) = PUMP INLET

* AS DETERMINED DURING SAFETY ANALYSIS

** CUBIC HORIZONTAL SUPPRESSOR MAY BE USED
WITH EITHER SUMP OUTLET

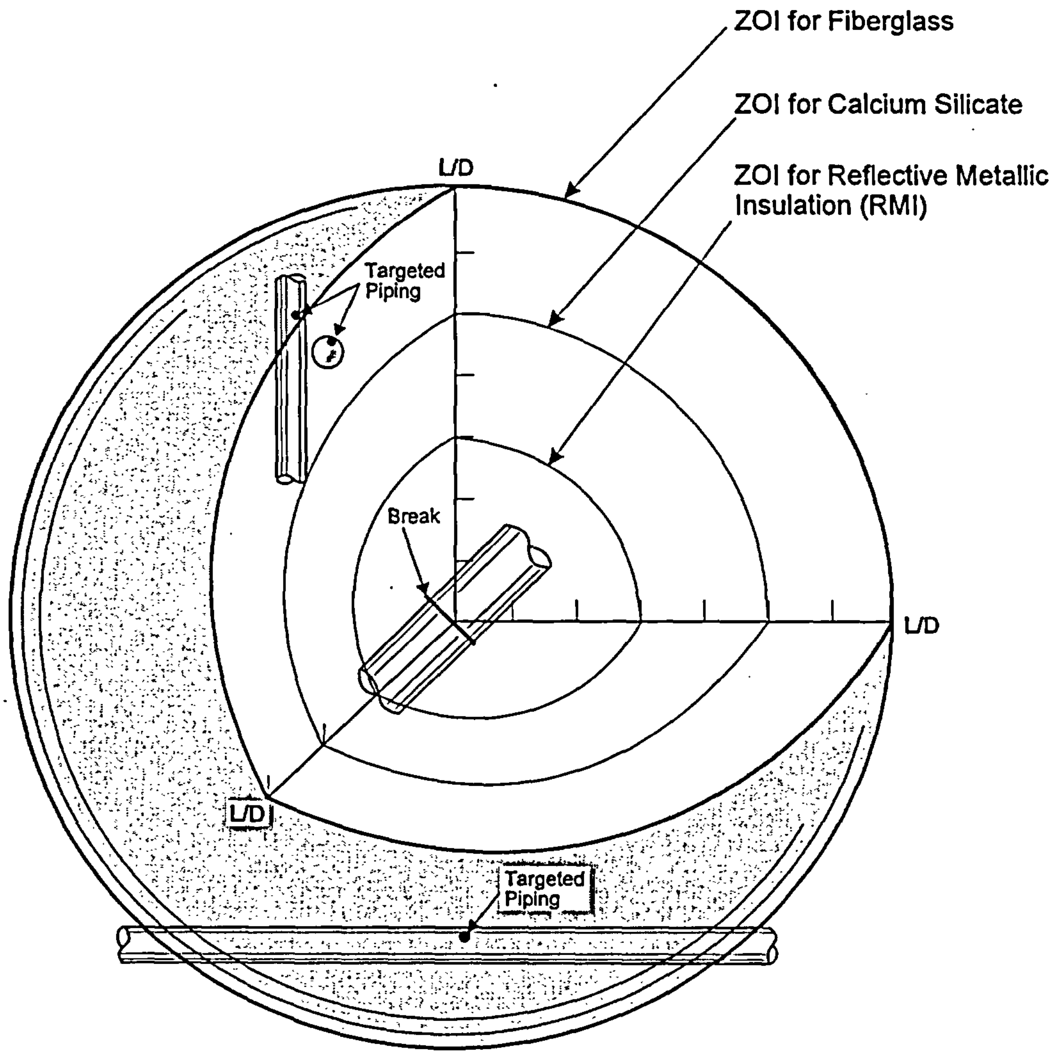
Note: Variations in Sump Features (e.g., Debris screen orientation and submergence level) exist, but not shown.

Figure 1. Conceptual Features of a PWR ECCS Recirculation Sump



Note: Variations in suppression pool features (e.g., number, design and location of suction screen and down comer vents) exist but are not shown.

Figure 2: Conceptual Features of a BWR Containment



Note:
L = Distance from break to target
D = Diameter of broken pipe

Figure 3. Zone of Influence (ZOI)

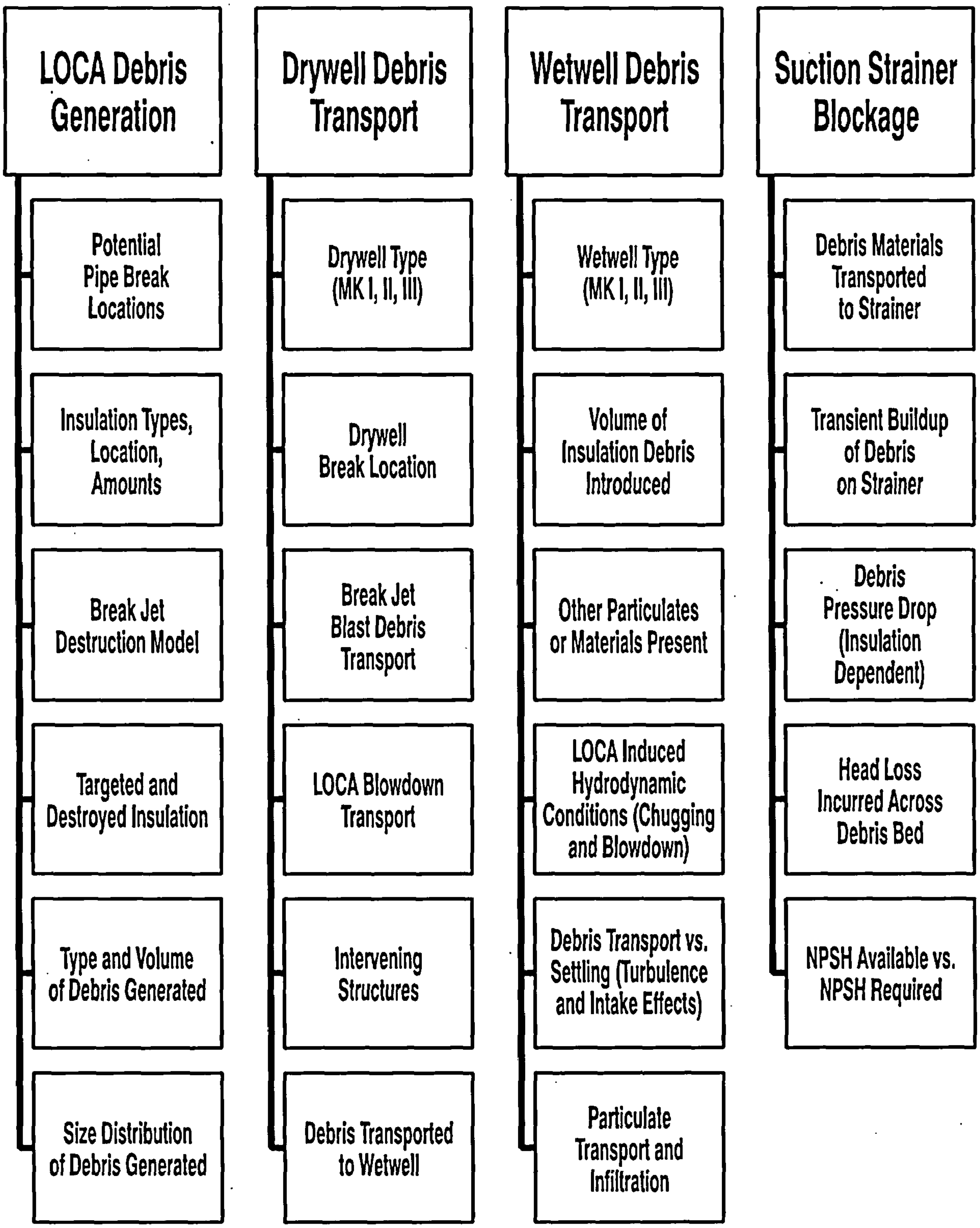


Figure 4. Debris Blockage Considerations for BWR LOCA Sequences

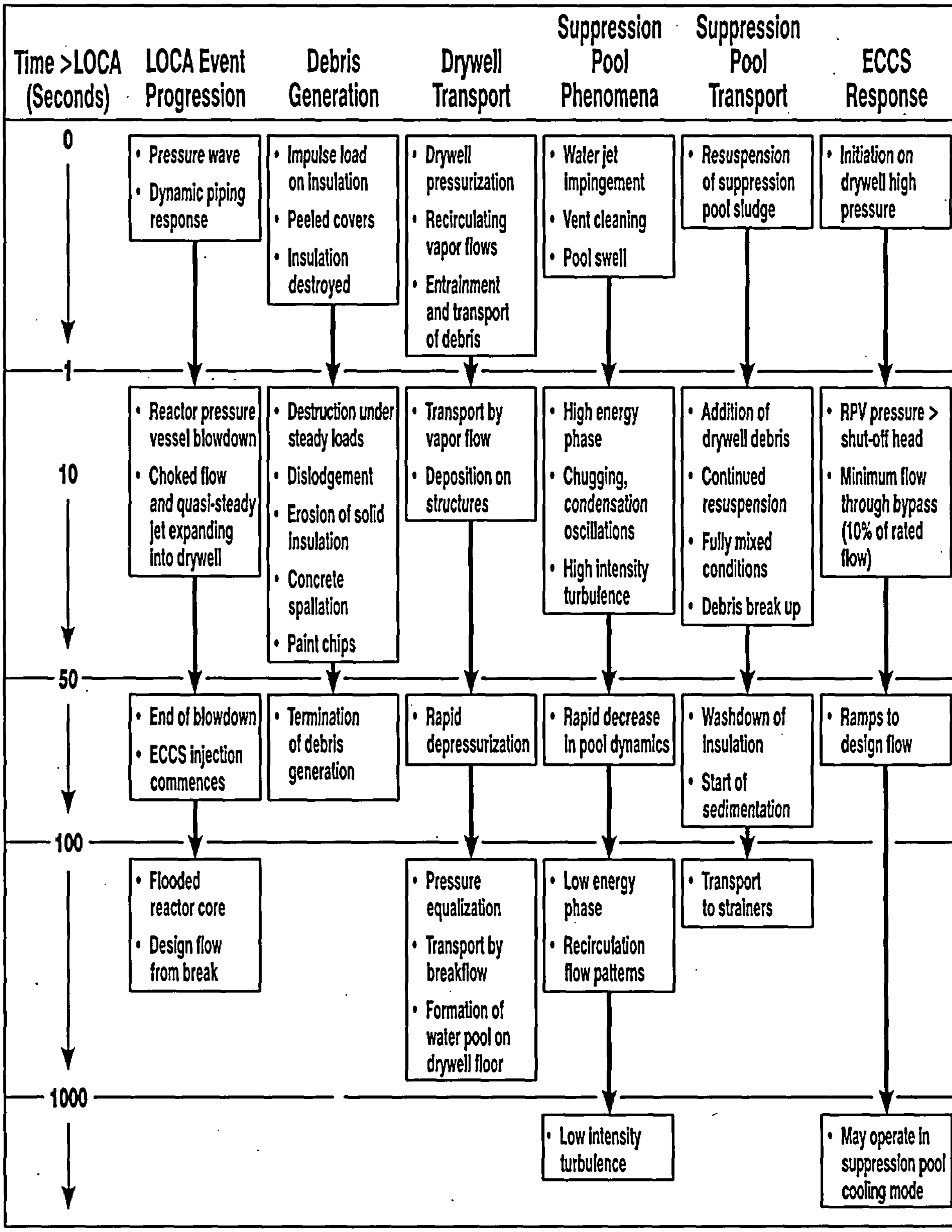


Figure 5. Events That May Effect Debris Blockage for BWR LOCA Sequences

APPENDIX A

GUIDELINES FOR REVIEW OF WATER SOURCES FOR EMERGENCY CORE COOLING

Water sources for long-term recirculation should be evaluated under possible post-LOCA conditions to determine the adequacy of their design for providing long-term recirculation. Technical evaluations can be subdivided into (1) sump hydraulic performance, (2) LOCA-induced debris effects, and (3) pump performance under adverse conditions. Specific considerations within these categories, and the combination thereof, are shown in Figure A-1. The final acceptance criterion is that adequate NPSH margin exists at the pump inlet under all postulated post-LOCA conditions.

SUMP HYDRAULIC PERFORMANCE

Sump hydraulic performance (with respect to air ingestion potential) can be evaluated on the basis of submergence level (or water depth above the PWR sump or BWR suction strainer outlets) and necessary pumping capacity (or pump inlet velocity). The water depth above the pipe centerlines and the inlet pipe velocity (U) can be expressed non-dimensionally as the Froude number:

$$\textit{Froude number} = \frac{U}{\sqrt{gs}}$$

where g is the acceleration due to gravity. Extensive experimental results have shown that the hydraulic performance of ECC sumps (particularly the potential for air ingestion) is a strong function of the Froude number. Other nondimensional parameters (e.g., Reynolds number and Weber number) are of secondary importance.

Sump hydraulic performance can be divided into three performance categories:

1. Zero air ingestion, for which vortex suppressors or increases in the required NPSH above that from the pump manufacturer's curves are not needed.
2. Air ingestion of 2% or less, a conservative level at which degradation of pumping capability is not expected based on an increase of the required NPSH.
3. Vortex suppressors to reduce air ingestion effects to zero.

For PWRs, zero air ingestion can be ensured by use of the design guidance set forth in Table A-l. Determination of those designs having ingestion levels of 2% or less can be obtained using correlations given in Table A-2 and the attendant sump geometric envelope. Geometric and screen guidelines for PWRs are contained in Tables A-3.1, A-3.2, A-4, and A-5. Table A-6 presents design guidelines for vortex suppressors that have shown the capability to reduce air ingestion to zero. These guidelines (Tables A-l through A-6) were developed from extensive hydraulic tests on full-scale sumps and provide a rapid means of assessing sump hydraulic performance. If the PWR

sump design deviates significantly from the bounding values of design parameters noted, similar performance data should be obtained for verification of adequate sump hydraulic performance.

For BWRs, full-scale tests of suppression pool suction strainer screen outlet designs for recirculation pumps have shown that air ingestion is zero for Froude numbers less than 0.8 with a minimum submergence of 6 feet, and operation up to a Froude number 1.0 with the same minimum submergence may be possible before air ingestion levels of 2% may occur (Revision 1 of NUREG-0897 and NUREG-2772).

LOCA-INDUCED DEBRIS EFFECTS

Assessment of LOCA debris generation and the determination of possible debris interceptor blockage is complex. The evaluation of this safety question is dependent on the types and quantities of insulation employed, the location of such insulation materials within containment and with respect to the sump or suppression pool strainer location, the estimation of quantities of debris generated by a pipe break, and the migration of such debris to the interceptors. Thus blockage estimates (i.e., generation, transport, and head loss) are specific to the insulation material, the piping layout, and the plant design.

Since break jet forces are the dominant debris generator, the predicted jet envelope will determine the quantities and types of insulation debris. Figure A-2 provides a conceptual three-region model that has been developed from analytical and experimental considerations as identified in Revision 1 of NUREG-0897 and NUREG/CR-6224. The destructive results (e.g., volume of insulation and other debris generated, size of debris) of the break jet forces will be considerably different for different types of insulation (Figure A-2), different types of installation methods, and distance from the break. Region I represents a total destruction zone; Region II represents a region where high levels of damage are possible depending on insulation type, whether encapsulation is employed, methods of attachment, etc.; and Region III represents a region where dislodgement of insulation in whole, or as-fabricated, segments is likely to occur. NUREG-0897 and NUREG/CR-6224 provide a more detailed discussion of these considerations. NUREG-0897, NUREG/CR-6224, NUREG/CR-2982, NUREG/CR-3170, NUREG/CR-3394, NUREG/CR-3616, NUREG/CR-6772, and NUREG/CR-6773 provide more detailed information relevant to assessing debris generation and transport.

PUMP PERFORMANCE UNDER ADVERSE CONDITIONS

The pump industry historically has determined required NPSH for pumps on the basis of a percentage degradation in pumping capacity. The percentage has at times been arbitrary, but generally is in the range of 1% to 3%. A 2% limit on allowed air ingestion is recommended since higher levels have been shown to initiate degradation of pumping capacity.

The 2% by volume limit on sump air ingestion and the NPSH criteria are applied independently. However, air ingestion levels less than 2% can also affect NPSH margin. If air ingestion is indicated, correct the required NPSH from the pump curves by the relationship:

$$NPSH_{required(\alpha p<2\%)} = NPSH_{required(liquid)} \times \beta$$

where $\beta = 1 + 0.50\alpha_p$ and α_p is the air ingestion rate (in percent by volume) at the pump inlet flange.

COMBINED EFFECTS

As shown in Figure A-l, three interdependent effects (i.e., sump or suction strainer performance, debris generation and transport, and pump operation under adverse conditions) warrant evaluation for determining long-term recirculation capability (i.e., loss of NPSH margin).

CRITERIA FOR EVALUATING SUMP FAILURE

The sump failure criterion depends on sump submergence and may be pump or system dependent. Figures A-3(a) and A-3(b) illustrate the two basic sump configurations of fully and partially submerged screens. Although only vertical sump configurations are shown here, the same designations are applicable to other screen designs. The key distinction between the fully and partially submerged configurations is that partially submerged screens allow equal pressure above both the pit and the pool, which are potentially separated by a debris bed. Fully submerged screens have a complete seal of water between the pump inlet and the containment atmosphere along all water paths passing through the sump screen. The effect of this difference in evaluation of the sump failure criterion is described below.

Fully Submerged Sump Screens

Figure A-3(a) presents a schematic of a fully submerged sump. The most likely mode of failure for sumps in this configuration is due to cavitation within the pump housing when head loss caused by debris accumulation exceeds the $NPSH_{Margin}$.* For this set of plants (in which sump screens are fully submerged at the time of switchover), the onset of cavitation is determined by comparing plant $NPSH_{Margin}$, which is part of the plant's licensing basis, with the screen head loss calculated in the plant evaluations performed per Regulatory Position 1.3. For this case, therefore, the sump failure criterion is assumed to be reached when

$$\text{Head Loss Across the Debris Bed} \geq NPSH_{Margin.}$$

Note that cavitation could occur in one pump housing while a different pump with a different NPSH margin may not have cavitation. Only in certain conditions (Regulatory Position 1.3.1.3) may credit be taken for continued operation under cavitating conditions, which could relax the above sump failure criterion for a brief period and provide an opportunity for recovery action.

* $NPSH_{Margin}$ is the amount by which NPSHA exceeds NPSHR. This definition is given in ANSI/HI 1.1-1.5 1994, "American National Standard for Centrifugal Pumps for Nomenclature, Definitions, Application and Operation," Section 1.3.3.1.16.2, "NPSH Margin Considerations."

The same standard, in Section 1.3.3.1.16, defines NPSHA and NPSHR.

- Net positive suction head available (NPSHA) is the total suction head of liquid absolute, determined at the first stage impeller datum, less the absolute vapor pressure of the liquid.
- Net positive suction head required (NPSHR) is the amount of suction head, over vapor pressure, required to prevent more than 3% loss in total head of the first stage of the pump at a specific capacity.

The head loss due to debris is not included in the definition of NPSHA. The value of the head loss due to debris is compared to the value of $NPSH_{Margin}$ in order to determine whether pump cavitation will occur.

Partially Submerged Sump Screens

Figure A-3(b) presents a schematic of a partially submerged sump. Failure can occur for sumps in this configuration in one of two ways: by pump cavitation as explained above or when head loss caused by debris buildup prevents sufficient water from entering the sump. This flow imbalance occurs when water infiltration through a debris bed on the screen can no longer satisfy the volumetric demands of the pump or pumps taking suction from the sump. Because the pit and the pool are at equal atmospheric overpressure, the only force available to move water through a debris bed is the static pressure head in the pool. Numeric simulations confirm that an effective head loss across a debris bed approximately equal to half the submerged screen height is sufficient to prevent adequate water flow, i.e., the force available to move water through the debris bed is approximately the average between the gravitational head at the full depth of the pool and zero head at the pool surface. For all partially submerged sump screens, the sump failure criterion is assumed to be reached when .

$$\text{Head Loss Across the Debris Bed} \geq NPSH_{Margin} \quad \text{or} \geq \tfrac{1}{2} \text{ of submerged screen height}$$

When this criterion is met, the water level on the downstream side of the screen would drop rapidly and all pumps taking suction from the sump would have insufficient flow for continued operation.

After switchover to ECCS recirculation, the sump configuration would likely change from partially submerged to fully submerged. This can occur for a number of reasons, including accumulation of containment-spray water, continued melting of ice-condenser reservoirs, and continued addition of the refueling water storage tank (RWST) inventory to the containment pool. As the pool depth changes during recirculation, the "wetted area" (or submerged area) of the sump screens can also change. The wetted area of the screen determines the average approach velocity of water that may carry debris, the accumulation of debris on the screen and subsequent head loss, and the gravitational head of the pool across the screen. The sump water level should be calculated as a function of time and a conservative assessment made of debris transport and accumulation on the sump screen. For systems such as the recirculation containment spray that could initiate suction from the recirculation sumps before ECCS switchover, the sump water level applicable at that time should be calculated.

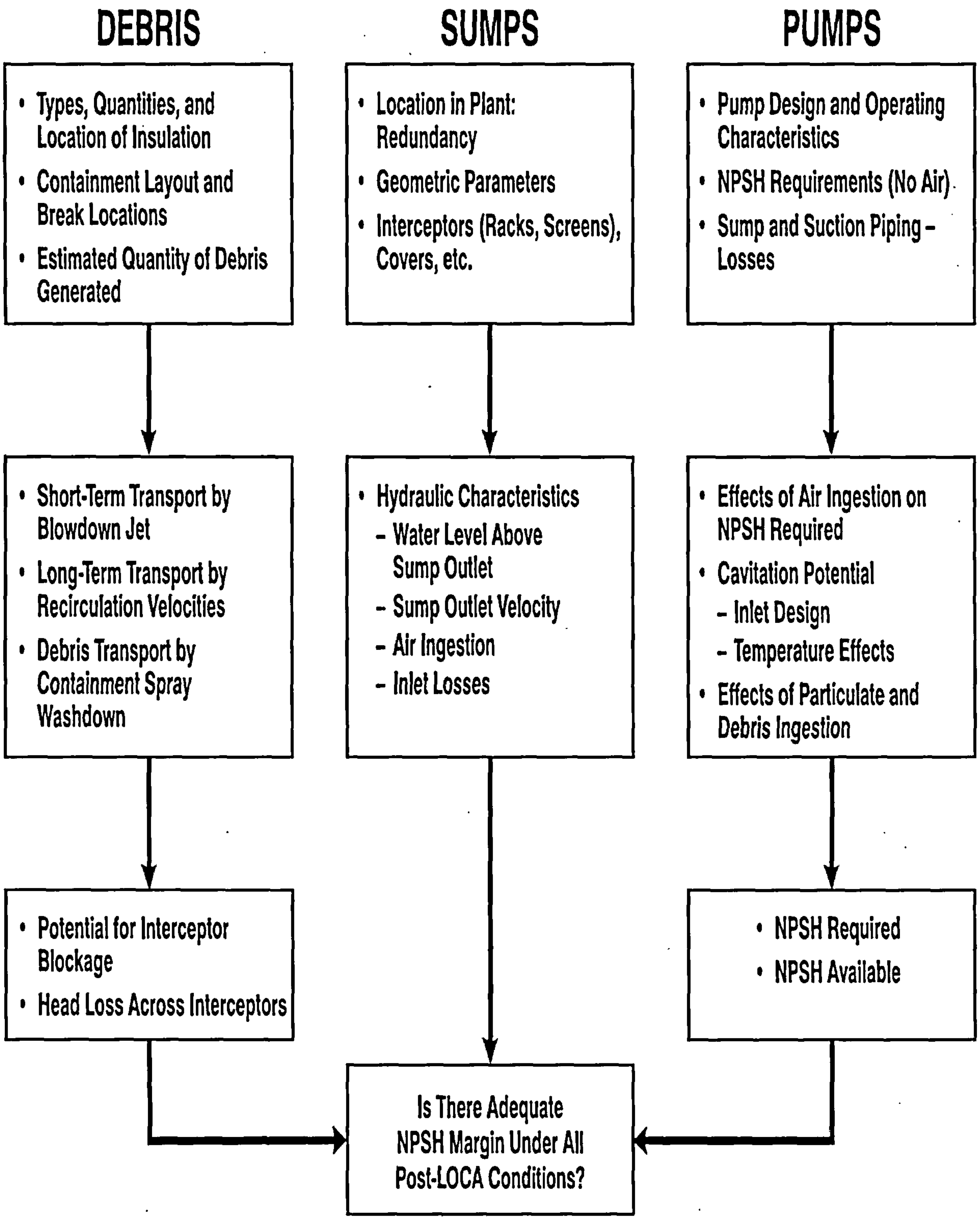


Figure A-1. Technical Considerations Relevant to PWR ECC Sump Performance

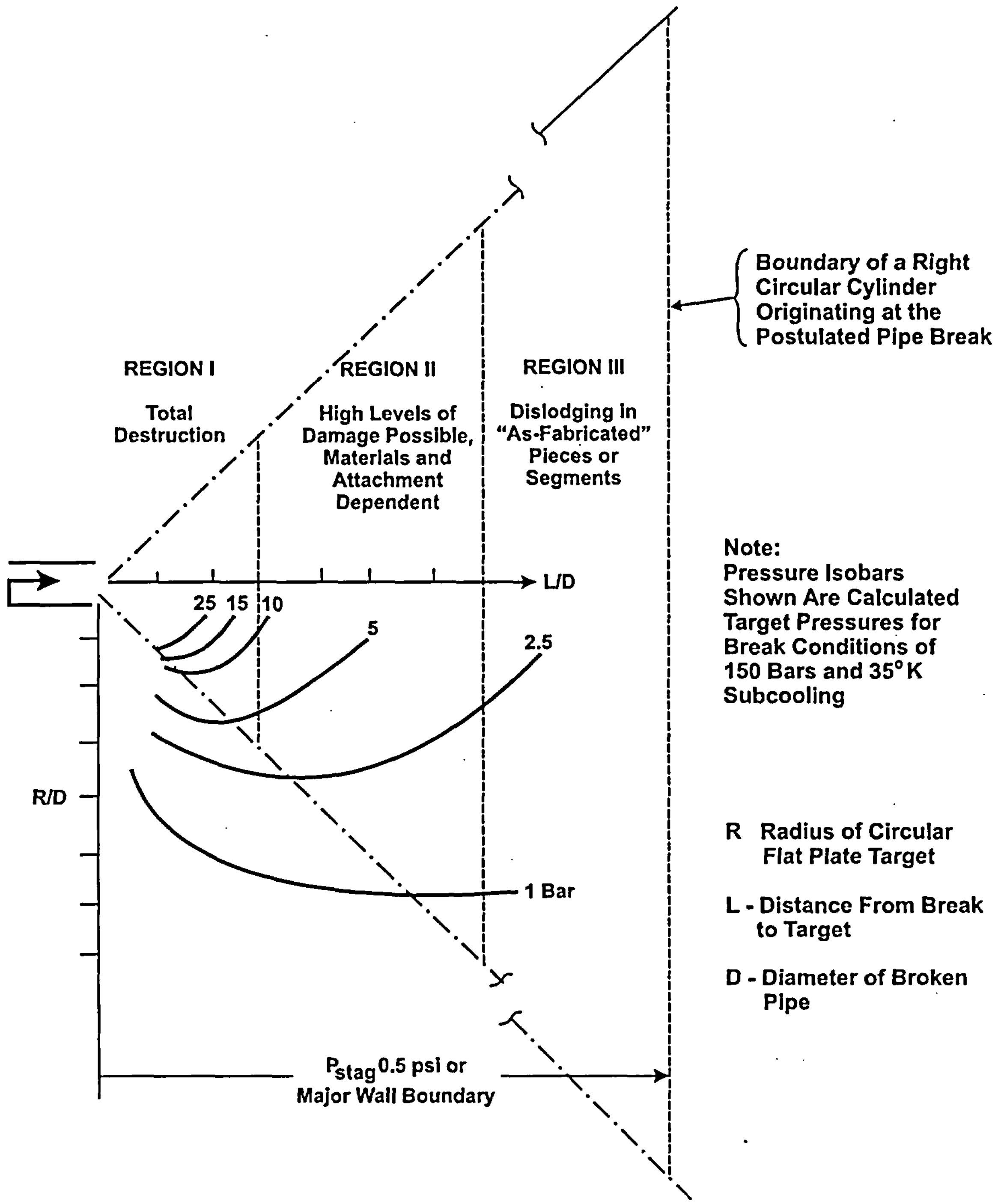
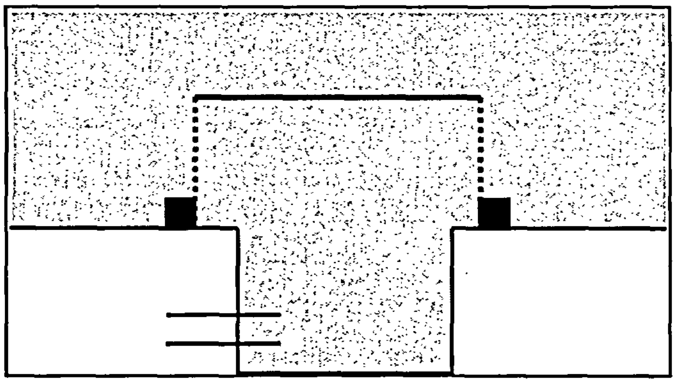
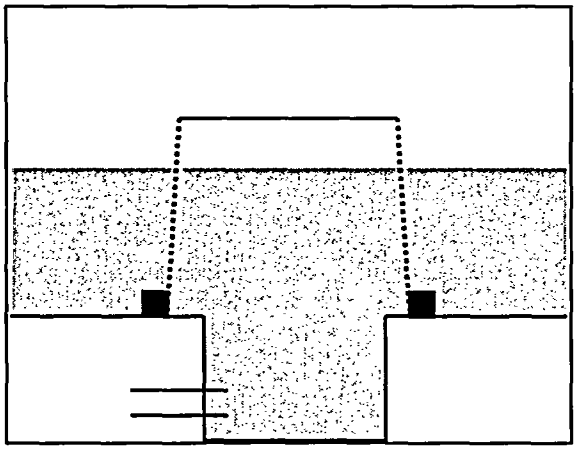


Figure A-2. Conceptual Multiple Region Insulation Debris Model



(a) Fully submerged screen configuration showing solid water from pump inlet to containment atmosphere.



(b) Partially submerged screen configuration showing containment atmosphere over both the external pool and the internal sump pit with water on lower portion of screen.

Figure A-3 Sump Screen Schematics

TABLE A-1

PWR HYDRAULIC DESIGN GUIDELINES FOR ZERO AIR INGESTION

Item	Horizontal Outlets	Vertical Outlets
Minimum Submergence, s (ft)	9	9
(m)	2.7	2.7
Maximum Froude Number, Fr	0.25	0.25
Maximum Pipe Velocity, U (ft/s)	4	4
(m/s)	1.2	1.2

NOTE: These guidelines were established using experimental results from NUREG/CR-2772, NUREG/CR-6224, and NUREG/CR-2982 and are based on sumps having a right rectangular shape.

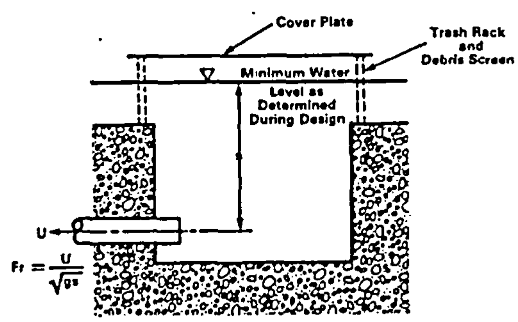


TABLE A-2

PWR HYDRAULIC DESIGN GUIDELINES FOR AIR INGESTION <2%

Air ingestion (α) is empirically calculated as
$\alpha = \alpha_o + (\alpha_1 \times Fr)$
where α_o and α_1 are coefficients derived from test
results as given in the table below

Item	Horizontal Outlets		Vertical Outlets	
	Dual	Single	Dual	Single
Coefficient α_o	-2.47	-4.75	-4.75	-9.14
Coefficient α_1	9.38	18.04	18.69	35.95
Minimum Submergence, s(ft)	7.5	8.0	7.5	10.0
(m)	2.3	2.4	2.3	3.1
Maximum Froude Number, Fr	0.5	0.4	0.4	0.3
Maximum Pipe Velocity, U(ft/s)	7.0	6.5	6.0	5.5
(m/s)	2.1	2.0	1.8	1.7
Maximum Screen Face Velocity (blocked and minimum submergence) (ft/s)	3.0	3.0	3.0	3.0
(m/s)	0.9	0.9	0.9	0.9
Maximum Approach Flow Velocity (ft/s)	0.36	0.36	0.3	0.36
(m/s)	0.11	0.11	0.11	0.11
Maximum Sump Outlet Coefficient, C_L	1.2	1.2	1.2	1.2

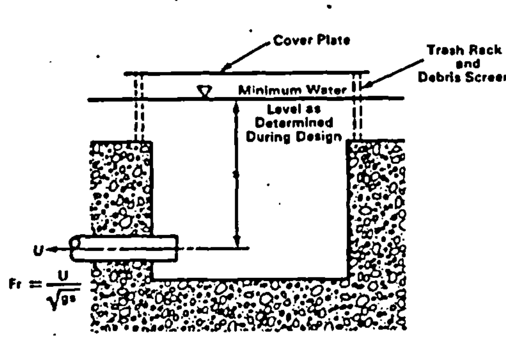


Table A-3.1

PWR GEOMETRIC DESIGN ENVELOP GUIDELINES FOR HORIZONTAL SUCTION OUTLETS

Sump Outlet	Sump Outlet Position*					
	e_y/d	$(B-e_y)/d$	c/d	b/d	f/d	e_x/d
Dual	>1	>3	>1.5	>1	>4	>1.5
Single					-	

* Preferred location.

Note: Dimensions are always measured to pipe centerline

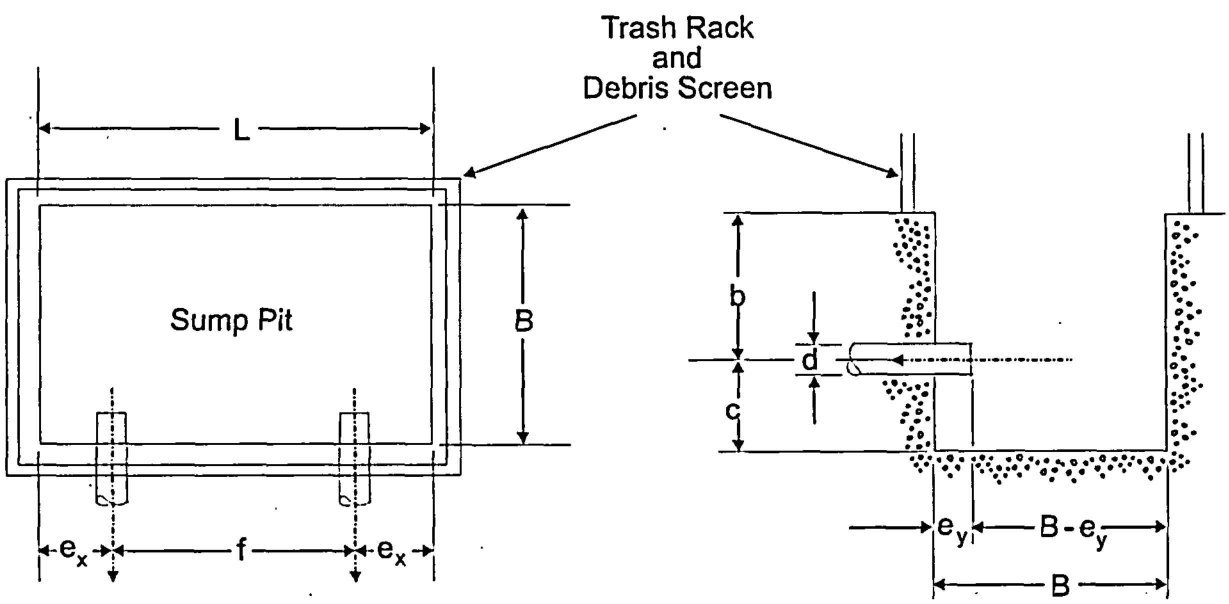


Table A-3.2

PWR GEOMETRIC DESIGN ENVELOP GUIDELINES FOR VERTICAL SUCTION OUTLETS

Sump Outlet	Sump Outlet Position*					
	e_y/d	$(B-e_y)/d$	c/d	b/d	f/d	e_x/d
Dual	>1	>1	>0	>1	>4	>1.5
Single	>1	>1	>1.5	>1	-	>1.5

* Preferred location.

Note: Dimensions are always measured to pipe centerline

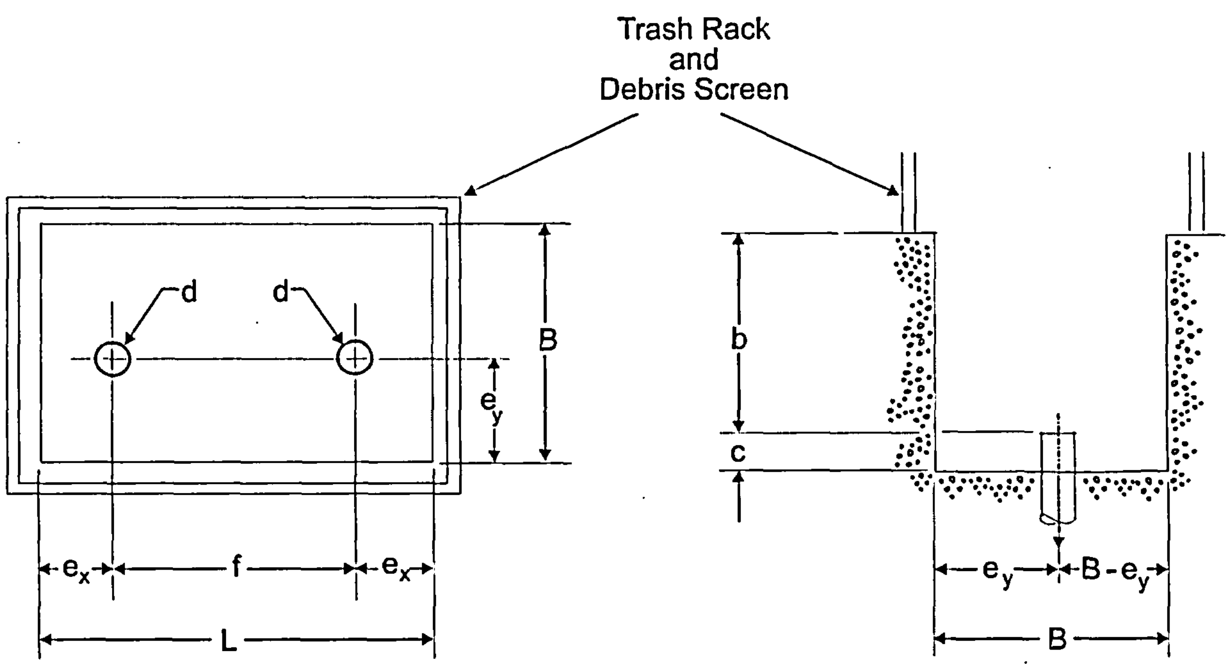


TABLE A-4

ADDITIONAL GUIDELINES RELATED TO SUMP SIZE AND PLACEMENT

1. The clearance between the trash rack and any wall or obstruction of length ℓ equal to or greater than the length of the adjacent screen/grate (B_s or L_s) should be at least 4 feet (1.2 meters).

2. A solid wall or large obstruction may form the boundary of the sump on one side only, i.e., the sump must have three sides open to the approach flow.

3. These additional guidelines should be followed to ensure the validity of the data in Tables A-1, A-2, A-3.1, and A-3.2.

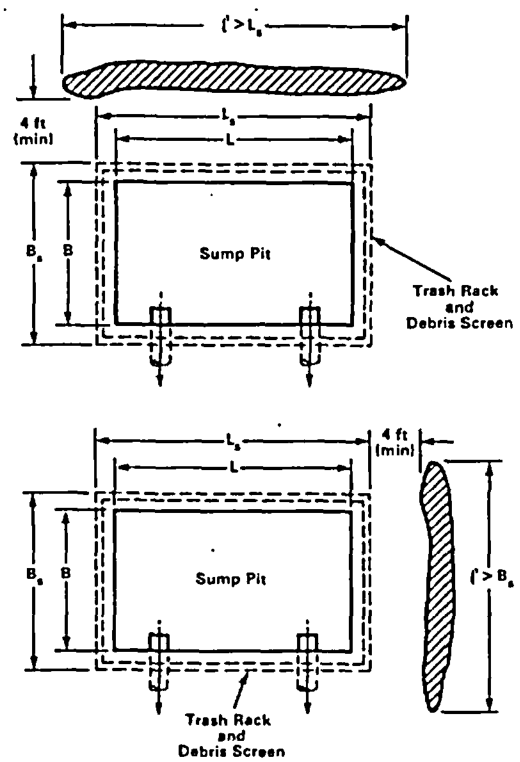


TABLE A-5

PWR DESIGN GUIDELINES FOR VERTICAL INTERCEPTORS AND COVER PLATE

1. Minimum height of interceptors should be 2 feet (0.61 meters).

2. Distance from sump side to screens, g_s, may be any reasonable value.

3. Screen mesh size (see Regulatory Position 1.1.1.12)

4. Trash racks should be vertically or nearly vertically oriented 1- to 1½-inch (25- to 38-mm) standard floor grate or equivalent.

5. The distance between the debris screens and trash racks should be 6 inches (15.2 cm) or less.

6. A solid cover plate should be mounted above the sump and should fully cover the trash rack. The cover plate should be designed to ensure the release of air trapped below the plate (a plate located below the minimum water level is preferable).

NOTE: See NUREG-0897.

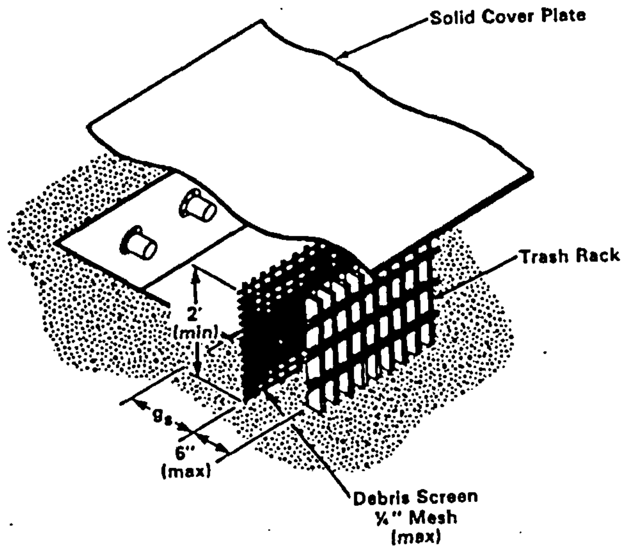


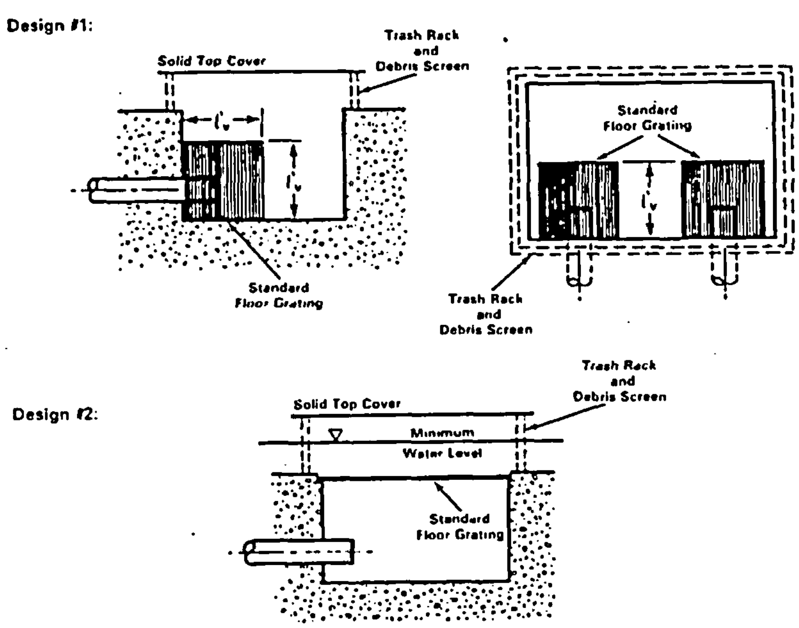
TABLE A-6

PWR GUIDELINES FOR SELECTED VORTEX SUPPRESSORS

1. Cubic arrangement of standard 1½-inch (30-mm) deep or deeper floor grating (or its equivalent) with a characteristic length, ℓ_v, that is at least 3 pipe diameters and with the top of the cube submerged at least 6 inches (15.2 cm) below the minimum water level. Noncubic designs with $\ell_v > 3$ pipe diameters for the horizontal upper grate and satisfying the depth and distances to the minimum water level given for cubic designs are acceptable.

2. Standard 1½-inch (38-mm) or deeper floor grating (or its equivalent) located horizontally over the entire sump and containment floor inside the screens and located below the lip of the sump pit.

NOTE: Tests on these types of vortex suppressors at Alden Research Laboratory have demonstrated their capability to reduce air ingestion to zero even under the most adverse conditions simulated.



APPENDIX A REFERENCES

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NUREG/CR-6773, "GSI-191: Integrated Debris Transport Tests in Water Using Simulated Containment Floor Geometries," USNRC, December 2002.[1]

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APPENDIX B
EXAMPLES OF ACTIVE MITIGATION SYSTEMS

In-Line (or Pipeline) Strainer

A strainer installed in the piping system, upstream of equipment, that will remove harmful objects and particulates from the fluid stream by a backwashing action.

Self-Cleaning Strainer

A strainer that is used upstream of equipment to filter out harmful objects and particulates and is designed to clean itself without the aid of external help.

Strainer Backwashing System

A system designed to dislodge objects and particulates from the surface of a strainer by directing a fluid stream in the opposite direction of the flow through the strainer.

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this proposed Revision 3 of Regulatory Guide 1.82 since the guide is being revised to clarify guidance for pressurized water reactors and to make minor changes to guidance for boiling water reactors; the guide continues to be intended to provide guidance on methods acceptable to the NRC staff for evaluating the adequacy of ECCS sump performance for recirculation cooling following a loss-of-coolant accident. Therefore, a new regulatory analysis is not needed.

In addition, the pertinent guidance in Regulatory Guide 1.1, also referred to as NRC Safety Guide 1, "Net Positive Suction Head For Emergency Core Cooling and Containment Heat Removal System Pumps," is incorporated into this Revision 3 of Regulatory Guide 1.82. Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997, requested licensees of nuclear power plants to respond to several questions related to the net positive suction head of the ECCS and containment heat removal system pumps in their power plants. The NRC staff reviewed these responses and wrote letters to the licensee of each power plant to provide the staff's conclusions based on these reviews. Based on its review of GL 97-04 responses, the staff determined that all operating plants satisfy the guidance in Safety Guide 1. The criteria used for these reviews were discussed in the generic letter and its regulatory analysis, in meetings with the NRC's Committee to Review Generic Requirements and Advisory Committee on Reactor Safeguards, and with licensees during the NRC's review of the generic letter responses. These criteria are now incorporated into this Revision 3 of Regulatory Guide 1.82. The portion of the guidance related to the use of containment pressure for the determination of available net positive suction head is taken from the guidance in Safety Guide 1.

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SAFETY GUIDE 1

NET POSITIVE SUCTION HEAD FOR EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SYSTEM PUMPS

A. Introduction

Proposed General Design Criterion 41 requires that the emergency cooling and containment heat removal systems be capable of accomplishing their required safety functions assuming partial loss of installed capacity. In current designs the ability to accomplish these safety functions reliably depends in part on the proper performance of system pumps which, in turn, depends on the conditions under which the pumps must operate. One of these conditions is suction pressure. This guide describes a suitable relationship between increases in containment pressure caused by postulated loss of coolant accidents and the net positive suction head (NPSH) of emergency core cooling and containment heat removal system pumps which may be used to implement General Design Criterion 41.

B. Discussion

A significant consideration related to emergency core cooling and containment heat removal systems is the potential for degraded pump performance which could be caused by a number of factors, including inadequate NPSH. If the NPSH available to a pump is not sufficient, cavitation of the pumped fluid can occur. This cavitation may reduce significantly the capability of the system to accomplish its safety functions.

It is important that the proper performance of emergency core cooling and containment heat removal systems be independent of calculated increases in containment pressure caused by postulated loss of coolant accidents in order to assure reliable operation under a variety of possible accident conditions. For example, if proper operation of the emergency core cooling system depends upon maintaining the containment pressure above a specified minimum amount, then too low an internal pressure (resulting from impaired containment integrity or operation of the containment heat removal systems at too high a rate) could significantly affect the ability of this system to accomplish its safety functions by causing pump cavitation. In addition, the deliberate continuation of a high containment pressure to maintain an adequate pump NPSH would result in greater leakage of fission products from the containment and higher potential offsite doses under accident conditions than would otherwise result.

Changes in NPSH for emergency core cooling and containment heat removal system pumps caused by increases in temperature of the pumped fluid under loss of coolant accident conditions can be accommodated without reliance on the calculated increase in containment pressure. Adequate NPSH can be assured by locating pumps at suitable elevations with respect to the storage volumes connected to their suction sides, by using multistage or booster pumps, by a combination of these methods, or by other techniques.

C. Regulatory Position

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 temparatures of pumped fluids and no increase in containment pressure from that present prior to postulated loss of coolant accidents.

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sel material should be continued in order to permit verification that expected material properties assure nonbrittle behavior of the reactor vessel throughout its lifetime under postulated accident conditions. It is expected that this determination can be made within 5 years.

2. During the 5-year period necessary to develop the needed data, the potential reactor pressure vessel thermal shock problem which may result from emergency core cooling system operation need not be reviewed in individual cases unless significant changes in presently approved core or reactor pressure vessel designs are proposed.
3. Should it be concluded that the margin of safety against reactor pressure vessel brittle failure due to emergency core cooling system operation at any time during vessel life is unacceptable, an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material. In the meantime, applicants should outline available engineering solutions and show that their designs do not preclude the use of such solutions.

June 1974

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.82

SUMPS FOR EMERGENCY CORE COOLING AND CONTAINMENT SPRAY SYSTEMS

A. INTRODUCTION

General Design Criteria 35, "Emergency Core Cooling," 36, "Inspection of Emergency Core Cooling System," 37, "Testing of Emergency Core Cooling System," 38, "Containment Heat Removal," 39, "Inspection of Containment Heat Removal System," and 40, "Testing of Containment Heat Removal System," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," require that a system be provided to remove the heat released to the containment following a postulated design basis accident (DBA) and that this system be designed to permit appropriate periodic inspection and testing to assure its integrity, capability, and operability. General Design Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50, requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This guide describes a method acceptable to the Regulatory staff for implementing these requirements with regard to design, fabrication, and testing of sump or suction inlet conditions for pumps in the emergency core cooling and containment spray systems. This guide applies to pressurized water reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

Sumps or pump intakes serve the emergency core cooling system (ECCS) and the containment spray system (CSS) by providing for collection of reactor coolant and chemically reactive spray solution and allowing its recirculation for additional cooling and fission product removal.

For optimum use of the available coolant, the sumps should be placed at the lowest level practical. There may be numerous places within the containment structure where coolant could accumulate during containment spray application, and these areas should be provided with drains or flow paths to the sump location to minimize coolant holdup in areas away from the sumps. This guide does not address design of the drains. Because a certain amount of debris may flow toward the sump, the drains entering the sump area should terminate in such a manner that the emerging flow would not tend to impinge upon the coolant sump.

The debris resulting from a loss-of-coolant accident (LOCA) may be divided into two categories: (1) the pieces that by virtue of weight and volume will tend to float or sink slowly and (2) the heavy pieces that will drop to the floor surface. Every effort should be made to prevent either category of debris from accumulating at the sump location. Because the small drainage sump for collecting and monitoring normal leakage within the containment is separate from the coolant sump intended to serve the ECCS and CSS pumps, the floor would normally slope down toward the drainage sump. These sumps for routine building drainage should be at a slightly lower elevation than the coolant sumps so that water from minor leaks and spills can not enter the ECCS-CSS sumps. The coolant sump location should be away from the drainage sump, so that the normal floor slope would assist in preventing heavier debris from accumulating at the coolant sump. In addition, the floor around the coolant sump should slope down and away from that sump to discourage debris from collecting on part of that sump structure.

Pump intakes should be protected by screens and trash racks (coarse outer screens) of sufficient strength

to resist impact loads that could be imposed by missiles that may be generated by the initial LOCA or by trash. Isolation of the coolant sump from high-energy pipe lines is an important consideration in missile protection. The screen and trash rack structures should be located above floor level to minimize the adverse effects from debris collecting on the screen structure. Redundant coolant sump screens and pump suction pipes should be separated as much as practical to reduce the possibility that a partially clogged screen or missile damage to one screen could adversely affect other pump circuits. In addition, the design of suction intakes should consider the avoidance of flow degradation by vortex formation.

It is expected that the water surface will be above the top of the screen structure after completion of the safety injection. However, the uncertainties about the extent of water coverage on the screen structure, the amount of floating debris that may accumulate, and the potential for early clogging do not favor the use of a horizontal top screen. Therefore, no credit should be taken in computation of the available surface area for any top horizontal screen, and the top of the screen structure should preferably be a solid deck.

Slowly settling debris which is small enough to pass through the trash rack openings could clog the inner screens if the coolant flow velocity is too great to permit the bulk of the debris to sink to the floor level. The inner screen should be vertically mounted to mininize settling of debris on the screen surface, and sufficient screen area should be provided to keep the coolant flow velocity at the screen approximately 6 cm/sec (0.2 ft/sec). Such a velocity will allow debris with a specific gravity of 1.05 or more to settle before reaching the screen surface.

Size of openings in the fine screens should be determined by the physical restrictions, including spray nozzles, that may exist in the systems which are supplied with coolant for the emergency sump. As a minimum, consideration should be given to building spray nozzles, coolant channel openings, and pump running clearances in sizing the fine screen. If the coolant channel openings in the core represent the smallest flow restriction, the minimum opening in the core channels which will allow design operation of the ECCS should be used in sizing the fine screen mesh size.

Consideration should also be given to partial screen blockage in sizing the fine screen in order to assure an adequate margin of conservatism on free flow area.

A significant consideration is the potential for degraded pump performance which could be caused by a number of factors, including net positive suction head (NPSH). If the NPSH available to a pump is not sufficient, cavitation may significantly reduce the capability of the system to accomplish its safety function. For the recommended design velocity at the fine inner screens considered in this guide, a negligible pressure drop is anticipated across the screens. The effect of partially blocked screens should be considered in the evaluation of the overall NPSH.

To assure the readiness and integrity of the rack and screens, access openings should be provided to permit inspection of the inside structures and pump suction inlet openings. Inservice inspection for trash racks, screens, and pump suction inlet openings should be performed on a regular basis at every refueling period downtime, and it should include visual examination for evidence of structural distress or corrosion. Inspection of the coolant sump components should be made late in the refueling program and thus help to assure the absence of construction debris in the coolant sump area. Any requirements for preoperational or periodic substantiation of adequate NPSH should be considered in the location and layout of the sump.

C. REGULATORY POSITION

Reactor building sumps which are designed to be a source of water for the emergency core cooling system (ECCS) and/or the containment spray system (CSS) following a loss-of-coolant accident (LOCA) should meet the following criteria:

1. A minimum of two sumps should be provided, each with sufficient capacity to serve one of the redundant halves of the ECCS and CSS systems.

2. The redundant sumps should be physically separated from each other and from high-energy piping systems by structural barriers, to the extent practical, to preclude damage to the sump intake filters by whipping pipes or high-velocity jets of water or steam.

3. The sumps should be located on the lowest floor elevation in the containment exclusive of the reactor vessel cavity. At a minimum, the sump intake should be protected by two screens: (1) an outer trash rack and (2) a fine inner screen. The sump screens should not be depressed below the floor elevation.

4. The floor level in the vicinity of the coolant sump location should slope gradually down away from the sump.

5. All drains from the upper regions of the reactor building should terminate in such a manner that direct streams of water, which may contain entrained debris, will not impinge on the filter assemblies.

6. A vertically mounted outer trash rack should be provided to prevent large debris from reaching the fine inner screen. The strength of the trash rack should be considered in protecting the inner screen from missiles and large debris.

7. A vertically mounted fine inner screen should be provided. The design coolant velocity at the inner screen should be approximately 6 cm/sec (0.2 ft/sec). The available surface area used in determining the design coolant velocity should be based on one-half of the free surface area of the fine inner screen to conservatively account for partial blockage. Only the vertical screens should be considered in determining available surface area.

8. A solid top deck is preferable, and the top deck should be designed to be fully submerged after a LOCA and completion of the safety injection.

9. The trash rack and screens should be designed to withstand the vibratory motion of seismic events without loss of structural integrity.

10. The size of openings in the fine screen should be based on the minimum restrictions found in systems served by the sump. The minimum restriction should take into account the overall operability of the system served.

11. Pump intake locations in the sump should be carefully considered to prevent degrading effects such as vortexing on the pump performance.

12. Materials for trash racks and screens should be selected to avoid degradation during periods of inactivity and operation and should have a low sensitivity to adverse effects such as stress-assisted corrosion that may be induced by the chemically reactive spray during LOCA conditions.

13. The trash rack and screen structure should include access openings to facilitate inspection of the structure and pump suction intake.

14. Inservice inspection requirements for coolant sump components (trash racks, screens, and pump suction inlets) should include the following:

a. Coolant sump components should be inspected during every refueling period downtime, and

b. The inspection should be a visual examination of the components for evidence of structural distress or corrosion.

ITEM A-43: CONTAINMENT EMERGENCY SUMP PERFORMANCE (REV. 1)

DESCRIPTION

Historical Background

This issue deals with a concern for the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water from the PWR containment sump, or the BWR RHR system suction intake, must be initiated and maintained to prevent core-melt. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired thereby seriously degrading long-term recirculation flow capability. The concern applies to both PWRs and BWRs. The RHR suction strainers in a BWR are analogous to the PWR sump debris screen and adequate recirculation cooling capacity is necessary to prevent core-melt following a postulated LOCA. The issue was declared a USI in January 1979 and published in NUREG-0510.[186]

The technical concerns evaluated under USI A-43 are as follows:

(1) PWR sump (or BWR RHR suction intake) hydraulic performance under post-LOCA adverse conditions resulting from potential vortex formation and air ingestion and subsequent pump failure.

(2) The possible transport of large quantities of LOCA-generated insulation debris resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling.

(3) The capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to possible air, debris, or other effects such as particulate ingestion on pump seal and bearing systems.

The staff's proposed resolution for USI A-43 was issued for public comment on May 10, 1983. The public comment package included NUREG-0869,[1056] the staff's technical findings report NUREG-0897,[1057] proposed Regulatory Guide 1.82, Revision 1, and proposed SRP[11] Section 6.2.2, Revision 4, "Containment Heat Removal Systems." A summary of the public comments received and the staff's response are contained in Appendix A of NUREG-0869,[1056] Revision 1.

CONCLUSION

In October 1985, the resolution of USI A-43 was presented to the Commission in SECY-85-349.[1060] The staff is implementing the resolution of USI A-43 through the following actions:

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(1) The staff's technical findings (NUREG-0897, Revision 1)[1057] were published for use as an information source by applicants, licensees, and the staff.

(2) SRP[11] Section 6.2.2 and Regulatory Guide 1.82[1058] were revised to reflect the staff's technical findings reported in NUREG-0897, Revision 1. This revised licensing guidance applies only to reviews of: (a) future construction permit applications and preliminary design approvals (PDAs); (b) final design approvals (FDAs) for standardized designs which are intended for referencing in future construction permit applications that have not received approval; and (c) applications for licenses to manufacture. This revised guidance became effective 6 months after issuance of Regulatory Guide 1.82, Revision 1.

(3) Generic Letter 85-22[1059] (for information only) was sent to all holders of an operating license or construction permit outlining the safety concerns regarding potential debris blockage and recirculation failure due to inadequate NPSH. It was recommended (but not required) that licensees utilize Regulatory Guide 1.82,[1058] Revision 1, as guidance for conduct of the 10 CFR 50.59 analysis for future plant modifications involving replacement of insulation on primary system piping and/or equipment. If, as a result of NRC staff review of licensee actions associated with replacement or modification to insulation, the staff decides that SRP[11] 6.2.2, Rev. 4 and/or Regulatory Guide 1.82,[1058] Rev. 1, criteria should be (or should have been) applied by the licensees, and the staff seeks to impose these criteria, then NRC will treat such actions as plant-specific backfits pursuant to 10 CFR 50.109.

Thus, this issue was RESOLVED and new requirements were established.

3150-0011

OMB No.

NRCB 96-03

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

May 6, 1996

NRC BULLETIN 96-03: POTENTIAL PLUGGING OF EMERGENCY CORE COOLING SUCTION
STRAINERS BY DEBRIS IN BOILING-WATER REACTORS

Addressees

All holders of operating licenses or construction permits for boiling-water reactors (BWRs), except Big Rock Point and holders of possession-only licenses.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to:

(1) request addressees to implement appropriate procedural measures and plant modifications to minimize the potential for clogging of emergency core cooling system (ECCS) suppression pool suction strainers by debris generated during a loss-of-coolant accident (LOCA), and

(2) require that addressees report to the NRC whether and to what extent the requested actions will be taken and to notify the NRC when actions associated with this bulletin are complete.

Background

On July 28, 1992, an event occurred at Barsebäck Unit 2, a Swedish BWR, which involved the plugging of two containment vessel spray system (CVSS) suction strainers. The strainers were plugged by mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that spuriously opened while the reactor was at 3,100 kPa [435 psig]. Two of the three strainers on the suction side of the CVSS pumps were in service and became partially plugged with mineral wool. Following an indication of high differential pressure across both suction strainers 70 minutes into the event, the operators shut down the CVSS pumps and backflushed the strainers. The Barseb„ck event demonstrated that the potential exists for a pipe break to generate insulation debris and

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transport a sufficient amount of the debris to the suppression pool to clog the ECCS strainers.

On January 16 and April 14, 1993, two events involving the clogging of ECCS strainers also occurred at the Perry Nuclear Power Plant, a domestic BWR. The first Perry event involved clogging of the suction strainers for the residual heat removal (RHR) pumps by debris in the suppression pool. The second Perry event involved the deposition of filter fibers on these strainers. The debris consisted of glass fibers from temporary drywell cooling unit filters that had been inadvertently dropped into the suppression pool, and corrosion products that had been filtered from the pool by the glass fibers which accumulated on the surface of the strainer. The Perry events demonstrated the deleterious effects on strainer pressure drop caused by the filtering of suppression pool particulates (corrosion products or "sludge") by fibrous glass materials entrained on the ECCS strainer surfaces. These corrosion products are typically present in varying quantities in domestic BWRs. The sludge is generated during normal operation, and the amount of sludge present in the pool depends on the frequency of pool cleanings/desludging conducted by the licensee. Separate test programs have been conducted by the Boiling Water Reactor Owners Group (BWROG) and the staff to quantify this filtering effect.

Based on these events, the NRC issued Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," on May 11, 1993. The bulletin requested licensees to remove fibrous air filters and other temporary sources of fibrous material, not designed to withstand a LOCA, from the containment. In addition, licensees were requested to take any immediate compensatory measures necessary to ensure the functional capability of the ECCS.

Following these events, the staff performed calculations to assess the vulnerability of each domestic BWR. The results of these calculations showed that the potential existed for the ECCS pumps to lose net positive suction head (NPSH) margin due to clogging of the suction strainers by LOCA-generated debris. The staff then conducted a detailed study of a reference BWR 4 plant with a Mark I containment. The preliminary results of the staff study are contained in a draft report, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," which was published in August 1994. The preliminary study results confirmed the results of the earlier staff calculations. The final version of this report was published as NUREG/CR-6224 in October 1995.

Members of the NRC staff also attended an Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) workshop on the Barsebäck incident held in Stockholm, Sweden, on January 26 and 27, 1994. Representatives from other countries at this conference discussed actions taken or planned which would prevent or mitigate the consequences of BWR strainer blockage. Based on the preliminary results of the staff's study, as reinforced by information learned at the OECD/NEA workshop, the staff issued NRC Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," on February 18, 1994. The purpose of the bulletin supplement was to request that BWR licensees take the

appropriate interim actions to ensure reliability of the ECCS so that the staff and industry would have sufficient time to develop a permanent resolution. In addition, the bulletin supplement informed licensees of pressurized-water reactors (PWRs) and BWRs of new information on the vulnerability of ECCS suction strainers in BWRs and containment sumps in PWRs to clogging during the recirculation phase of a LOCA.

On September 11, 1995, Limerick Unit 1 was being operated at 100-percent power when control room personnel observed alarms and other indications that one safety relief valve (SRV) was open. Emergency procedures were implemented. Attempts to close the valve were unsuccessful, and a manual reactor scram was initiated. Prior to the opening of the SRV, the licensee had been running the "A" loop of suppression pool cooling to remove heat being released into the pool by leaking SRVs. Shortly after the manual scram, and with the SRV still open, the "B" loop of suppression pool cooling was started. Operators continued working to close the SRV and reduce the cooldown rate of the reactor vessel. Approximately 30 minutes later, fluctuating motor current and flow were observed on the "A" loop. Cavitation was believed to be the cause, and the loop was secured. After it was checked, the "A" pump was successfully restarted and no further problems were observed.

After the cooldown following the blowdown event, a diver was sent into the suppression pool at Unit 1 to inspect the condition of the strainers and the general cleanliness of the pool. Both suction strainers in the "A" loop of suppression pool cooling were found to be almost entirely covered with a thin "mat" of material, consisting mostly of fibers and sludge. The "B" loop suction strainers had a similar covering, but less of it. Analysis showed that the sludge was primarily iron oxides and the fibers were polymeric in nature. The source of the fibers was not positively identified, but the licensee has determined that the fibers did not originate within the suppression pool, and that no trace of either fiberglass or asbestos was in the fibers.

The Limerick event demonstrated the need to ensure adequate suppression pool cleanliness. In addition, it re-emphasized that materials other than fibrous insulation could also clog strainers (Perry's strainers were clogged by fibrous filter media). In response to this event, the staff issued NRC Bulletin 95-02, "Unexpected Clogging of Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," on October 17, 1995. The bulletin requested that licensees (1) assess the operability of their ECCS based on the cleanliness of their suppression pool and ECCS strainers, (2) verify the operability of the ECCS through an appropriate pump test and strainer inspection within 120 days from the date of the bulletin, (3) establish a pool cleaning program, (4) review their foreign material exclusion practices and correct any identified weaknesses, and (5) implement any appropriate additional measures for ensuring the availability of their ECCS. The staff is still reviewing the responses to NRC Bulletin 95-02, but results of the review of requested action (1) have shown that almost all plants have cleaned their pools during the last 4 years with most having done so during their last refueling outage.

Licensee responses to NRC Bulletin 93-02 and its supplement have demonstrated that appropriate interim measures have been implemented by licensees to ensure adequate protection of public health and safety, and to allow continued operation until the final actions requested in this bulletin are implemented. In responding to these bulletins, licensees ensured that (1) alternate water sources (both safety and nonsafety-related sources) to mitigate a strainer clogging event were available, (2) emergency operating procedures (EOPs) provided adequate guidance on mitigating a strainer clogging event, (3) operators were adequately trained to mitigate a strainer clogging event, and (4) loose and temporary fibrous materials stored in containment were removed. Licensee responses to NRC Bulletin 95-02 have shown that most suppression pools have been cleaned recently, and that those licensees who have not cleaned their suppression pools recently are scheduled to do so during their upcoming refueling outage. In addition, a generic safety assessment conducted by the BWROG concluded that operators would have adequate time to make use of alternate water sources (25-35 minutes). The staff also notes that the probability of the initiating event is low. The actions requested in this bulletin will ensure that the ECCS can perform its safety function and minimize the need for operator action to mitigate a LOCA.

Discussion

The results of the staff study, documented in NUREG/CR-6224, demonstrate that for the reference plant, there is a high probability that the available NPSH margin for the ECCS pumps will be inadequate following dislodging of insula- tion and other debris caused by a LOCA and transport of the debris to the suction strainers. In addition, the study calculated that the loss of NPSH could occur quickly (less than 10 minutes into the event). The study also demonstrated that determining the adequacy of NPSH margin for an ECCS system is highly plant-specific because of the large variations in such plant characteristics as containment type, ECCS flow rates, insulation types, plant layout, plant cleanliness, and available NPSH margin.

The Barsebäck event demonstrated that a pipe break can generate and transport sufficient quantities of insulation and other debris to the suppression pool where they can be potentially deposited onto strainer surfaces and cause the ECCS to lose NPSH. The Perry events further demonstrated that fibrous debris combined with corrosion products present in the suppression pool (sludge) can exacerbate the problem. This phenomenon was confirmed in the staff study which showed that the calculated loss of NPSH could occur soon (less than 10 minutes) after ECCS initiation. The effect of filtering sludge from the suppression pool water by fibrous debris deposited on the strainer surface was further confirmed in NRC-sponsored testing conducted at the Alden Research Laboratory which demonstrated that the pressure drop across the strainer was greatly increased by this filtering effect. Additional testing sponsored by the NRC at Alden Research Laboratory demonstrated that the energy conveyed to the suppression pool during the "chugging" phase of a LOCA is sufficient to ensure that the fibrous debris and sludge are well mixed and evenly distributed in the suppression pool, and can remain suspended for a sufficiently long period to allow large quantities to be deposited onto the strainer surfaces. The staff has concluded that this problem is applicable to all domestic BWRs.

The basis for the staff's conclusion is as follows: (1) there do not appear to be any features specific to a particular plant, class of plants, or containment type that would mitigate or prevent the generation, the transport to the suppression pool, or the deposition on the ECCS strainers of sufficient material to clog the strainers, and (2) parametric analyses performed in support of the NUREG/CR-6224 study, using parameter ranges which bound most domestic BWRs, failed to find parameter ranges that would prevent BWRs with other containment types from being susceptible to this problem. In addition, the staff study was conducted on a Mark I; Barsebäck had a strainer clogging event and is similar in design to a Mark II; and Perry, a Mark III, also had a strainer clogging event.

Section 50.46 of Title 10 of the Code of Federal Regulations (10 CFR 50.46) requires that licensees design their ECCS systems to meet five criteria, one of which is to provide long-term cooling capability of sufficient duration following a successful system initiation so that the core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. The ECCS is designed to meet this criterion, assuming the worst single failure. Experience gained from operating events and detailed analysis, as previously discussed, demonstrate that excessive buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers is highly likely to occur, creating the potential for a common-cause failure of the ECCS, which could prevent the ECCS from providing long-term cooling following a LOCA. The staff concludes therefore, that this issue must be resolved by licensees in order to ensure compliance with the regulations. Regulatory Guide 1.82, Revision 2 (RG 1.82, Revision 2), "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," provides an acceptable method of ensuring compliance with 10 CFR 50.46.

Plant-specific analyses to resolve this issue are difficult to perform because a substantial number of uncertainties are involved. Examples of these uncertainties include the amount of debris that would be generated by a pipe break for various insulation types; the amount of debris that would be transported to the suppression pool; the characteristics of debris reaching the suppression pool (e.g., size and shape); and head-loss correlations for various insulation types combined with suppression pool corrosion products, paint chips, dirt, and other particulates. Many of these uncertainties would be plant-specific because of the differences in plant characteristics such as plant layout, insulation types, ECCS flow rates, containment types, plant cleanliness, and NPSH margin. Testing may be required to quantify these uncertainties for licensees to demonstrate compliance with 10 CFR 50.46.

The staff has also closely followed the work of the BWROG to resolve this issue. The BWROG has evaluated several potential solutions, and has completed testing on three new strainer designs: two passive strainer designs and one self-cleaning design. The ongoing BWROG effort is consistent with the options proposed in this bulletin for resolution of the ECCS potential strainer clogging issue. These options are discussed in the next section under Requested Actions. The BWROG is also developing a utility resolution guidance (URG) document for providing the utilities with (1) guidance on

evaluation of the ECCS potential strainer clogging issue for their plant, (2) a standard industry approach to resolution of the issue that is technically sound, and (3) guidance that is consistent with the requested actions in this bulletin for demonstrating compliance with 10 CFR 50.46. The URG will include guidance on a calculational methodology for performing plant specific evaluations. This methodology is still under development by the BWROG. The staff considers the URG to be an important part of the implementation of the final resolution of this issue, and will closely monitor its development and application.

The staff has noted that much of the effort and discussion on this issue to date has focused on the threat caused by fibrous insulation. The staff recognizes that fibrous insulation represents the largest source of fibrous material in the containment; however, licensees are reminded that both the Perry and the Limerick events involved other sources of fibrous debris. In determining their resolution for this issue, licensees should focus on protecting the functional capability of the ECCS from all potential strainer clogging mechanisms.

Requested Actions

All BWR licensees are requested to implement appropriate measures to ensure the capability of the ECCS to perform its safety function following a LOCA. The staff has identified three potential resolution options; however, licensees may propose others which provide an equivalent level of assurance that the ECCS will be able to perform its safety function following a LOCA. The three options identified by the staff are as follows:

Option 1: Installation of a large capacity passive strainer design.

If this option is selected by a licensee, the strainer design used should have sufficient capacity to ensure that debris loadings equivalent to a scenario calculated in accordance with Section C.2.2 of RG 1.82, Revision 2, do not cause a loss of NPSH for the ECCS. This option has two main advantages. First, it is completely passive and, therefore, requires no operator intervention. Second, it does not require an interruption of ECCS flow. While this is the most advantageous of the options identified, the staff recognizes that it may be difficult for some licensees to implement this option owing to the difficulty in providing sufficient structural support for the strainers to handle LOCA-induced hydrodynamic loads. However, the staff notes that licensees may take appropriate measures in combination with this option to reduce the potential debris sources in containment and the suppression pool, which would, in turn, reduce the required capacity and physical size of the strainer, and therefore, assist in reducing the structural burden of the strainer installation. Licensees choosing this option for resolution should establish new or modify existing programs, as necessary, to ensure that the potential for debris to be generated and transported to the strainer surface does not at any time exceed the assumptions used in estimating the amounts of debris for sizing of the strainers in accordance with RG 1.82, Revision 2.

Option 2: Installation of a self-cleaning strainer.

This option automatically prevents strainer clogging by providing continuous cleaning of the strainer surface with a scraper blade or brush. Like Option 1, the self-cleaning strainer design would not rely on operator action or interrupt ECCS flow. However, this option does rely on an active component which is fully exposed to the LOCA effects in the suppression pool to keep the strainer surface clean. Therefore, appropriate measures should be taken to ensure the operability of the strainer. Installation of this type of strainer should be combined with the following measures to protect the strainer and ensure its operability: (1) implementation of reasonable measures to eliminate debris sources that could potentially damage or overload the strainer during a LOCA, including, as a minimum, removal of all debris from the suppression pool every refueling outage, and (2) implementation of surveillances to ensure adequate cleaning of the suppression pool and the operability of the strainer.

Option 3: Installation of a backflush system.

The backflush system is a reactive system that relies on operator action to remove debris from the surface of the strainer to prevent it from clogging. In order to ensure that operators can adequately deal with a strainer clogging event, installation of this type of system should be combined with the following measures: (1) reasonable measures to maximize the amount of time before clogging could occur; (2) instrumentation and alarms to indicate when strainer differential pressure increases; (3) operator training on recognition and mitigation of a strainer clogging event; and (4) implementation of surveillances to ensure the operability of the strainer instrumentation and backflush system. A supporting analysis for installation of a backflush system that is consistent with Section C.2.2 of RG 1.82 Revision 2 should be performed to demonstrate that operators have sufficient time to recognize the onset of clogging and to take appropriate action, taking into consideration their other responsibilities after a LOCA. In addition, this analysis should ensure that operators have the capability and sufficient time to cycle backflushing at the expected frequency and for the required total number of actuations anticipated in providing long-term core cooling following a LOCA. The suction strainers and backflush system should be so designed that interruption of ECCS flow due to backflushing during an accident does not contradict the guidance provided in the plant emergency operating procedures (EOPs). For instance, if the EOPs indicate that all available pumps should be running and injecting into the vessel, the system should be designed to ensure that interruption of ECCS flow for backflushing is not required during this stage of the accident. If EOPs indicate that unnecessary pumps may be secured, then use of backflush on the suction strainers of the unnecessary pumps would be acceptable.

The staff considers the instrumentation (e.g., strainer pressure differential or pump flow rate) relied upon by operators to indicate when a manual initiation of the strainer backflush system is required to be Type A instrumentation as defined in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following An Accident," Revision 3. The instrumentation should therefore be included with other Type A instrumentation in the

appropriate section of the technical specifications. In NUREG-1433, "Standard Technical Specfications, General Electric Plants, BWR/4," and NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6," (Volume 1, Revision 1) the applicable section is Section 3.3.3.1-1 "Post Accident Monitoring Instrumentation." The licensee should also provide appropriate corresponding bases.

Any components or systems installed to ensure that the ECCS can perform its safety function during a LOCA are considered by the staff to be a part of the ECCS. Therefore, these components or systems should be designed, fabricated, and tested to the same standards as the ECCS. Any request to deviate from this position would require an exemption with a supporting technical analysis and must meet the specific requirements of 10 CFR 50.12. Active features such as backflush and the self-cleaning strainer must be supported by test data that demonstrate the design effectiveness for removal of debris entrained on the surface of the strainer. Strainers installed for Option 1 must be supported by test data that demonstrate their performance characteristics and their ability to handle the worst case scenario for debris deposition on the strainer surface.

Section 50.36 of Title 10 of the Code of Federal Regulations (10 CFR 50.36) has been amended to provide the criteria for determining the content of Technical Specifications (TS) for nuclear power reactors. The amended rule was published in the Federal Register on July 19, 1995 (60 FR 36953). Paragraph (c)(2)(ii) of 10 CFR 50.36 provides four criteria for determining if a limiting condition for operation (LCO) is required in the TS. Criterion 3 states that a "structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier" should have a LCO in the TS. The staff believes that passive strainers, self-cleaning strainers, and strainer backflush systems meet Criterion 3 of the Commission's regulations and should be included in the TS because these components are necessary for the primary success path (i.e., the ECCS) to mitigate a design basis LOCA. However, since strainers and backflush systems are fundamental parts of the ECCS, the staff has concluded that the addition of new LCOs and action statements are not necessary. Rather, the effect of one of these components or systems being inoperable should be analyzed for its effect on the operability of the ECCS as a whole, and the appropriate ECCS action statement entered as a result. TS should be proposed to support surveillances for components and systems installed in response to this bulletin and should include, where appropriate, for the option selected, surveillance testing of active features (i.e., Options 2 and 3), and visual inspections where they provide reasonable assurance that the component is operable. Where appropriate, these TS surveillances should be proposed for existing strainer components to ensure their operability if a licensee determines that no modification to their ECCS strainers is necessary in response to this bulletin. Attachment 1 to this bulletin provides sample TS surveillances that are consistent with the format for the standard TS for the BWR 4, which may be used by licensees in determining appropriate TS surveillances for the actions implemented in response to this bulletin. Success criteria for the surveil- lances should be defined by the licensee in the bases section of the TS.

Plant procedures and other actions implemented in response to NRC Bulletin 93-02 and its supplement should remain in place until the final corrective actions requested in this bulletin have been implemented.

All licensees are requested to implement these actions by the end of the first refueling outage starting after January 1, 1997. This timeframe for implementation of the final resolution is considered appropriate by the staff owing to the interim actions already taken by licensees and the low probability of the initiating event.

Required Response

All addressees are required to submit the following written reports:

(1) Within 180 days of the date of this bulletin, a report indicating whether the addressee intends to comply with these requested actions, including a description of planned actions and mitigative strategies to be used, the schedule for implementation, and proposed TS, if appropriate; or, if the licensee does not intend to comply with these actions, a detailed description of the safety basis for the decision. The report must contain a detailed description of any proposed alternative course of action, the schedule for completing this alternative course of action, the safety basis for determining the acceptability of the planned alternative course of action, and proposed TS, if appropriate, that support the proposed alternative course of action and are consistent with 10 CFR 50.36. The staff considers the 180-day response period appropriate, given the amount of engineering that licensees may wish to perform before they provide their formal response to the staff.

(2) Within 30 days of completion of all requested actions, a report confirming completion and summarizing any actions taken.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy of the reports to the appropriate regional administrator.

Related Generic Communications

NRC Bulletin 95-02, "Unexpected Clogging of Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995.

NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993, and its supplement, dated February 18, 1994.

Backfit Discussion

The actions requested by this bulletin are considered backfits in accordance with NRC procedures and are necessary to ensure that licensees are in compliance with existing NRC rules and regulations. Specifically, 10 CFR 50.46 requires that adequate ECCS flow be provided to maintain the core temperature at an acceptably low value and to remove decay heat for the extended period of time required by the long-lived radioactivity remaining in the core following a design-basis accident. Therefore, this bulletin is being issued as a compliance backfit under the terms of 10 CFR 50.109(a)(4)(i), and a full backfit analysis was not performed. An evaluation was performed in accordance with NRC procedures, including a statement of the objectives and the reasons for the requested actions and the basis for invoking the compliance exception. A copy of this evaluation will be made available in the NRC Public Document Room.

Paperwork Reduction Act Statement

This bulletin contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0011, which expires July 31, 1997.

The public reporting burden for this collection of information is estimated to average 160 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the collection of information contained in the bulletin and on the following issues:

(1) Is the proposed collection of information necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

(2) Is the estimate of burden accurate?

(3) Is there a way to enhance the quality, utility, and clarity of the information to be collected?

(4) How can the burden of the collection of information be minimized, including the use of automated collection techniques?

Send comments on any aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, T-6 F33, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

signed by

Brian K. Grimes, Acting Director
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Management
Office of Nuclear Reactor
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Attachments:
1. Sample Technical Specification Surveillances

SAMPLE TECHNICAL SPECIFICATION SURVEILLANCES

SURVEILLANCE
FREQUENCY

SR 3.5.1.13

(a) Verify, by visual inspection, that each ECCS [18] months
suction strainer is not restricted by debris, that the supporting structure shows no evidence of structural distress or abnormal corrosion, and there is no evidence of abnormalities which could affect the mechanical functioning of the suction strainer.

(b) Verify suppression pool is adequately clean [18] months

[SR 3.5.1.14

(a) Verify that each [Self-Cleaning Strainer] attains at least [ ] rpm with a pressure differential of less than or equal to [ ] while the ECCS pump[s], taking suction from the strainer, is producing a flow rate of at least [ ] gpm.] [18] months

[SR 3.5.1.15

Verify that the [Strainer Backflush System] attains flow rate of at least [ ] gpm at each ECCS strainer.] [18] months

ISSUE 191: ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE (REV. 1)

DESCRIPTION

Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of Issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings. Thus, this issue was identified[1691] by NRR and called for an expanded research effort to address these new safety concerns.

CONCLUSION

A study was deemed to be required to determine whether PWR ECCS sumps are adequate to ensure proper ECCS operation. Based on the existence of an action plan[1692] to address the safety concerns, the issue was considered nearly-resolved in September 1996. It was later given a HIGH priority ranking in SECY-98-166.[1718]

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DPS Exhibit #7
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OMB Control No.: 3150-0012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555

June 9, 2003

NRC BULLETIN 2003-01: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to:

(1) Inform addressees of the results of NRC-sponsored research identifying the potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage in the event of a high-energy line break (HELB) requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS).

(2) Inform addressees of the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

(3) Request that, in light of these potentially adverse effects, addressees confirm their compliance with 10 CFR 50.46(b)(5) and other existing applicable regulatory requirements, or describe any compensatory measures implemented to reduce the potential risk due to post-accident debris blockage as evaluations to determine compliance proceed.

(4) Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Background

In 1979, as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." To support the resolution of USI A-43, the NRC undertook an extensive research program, the technical findings of which are summarized in NUREG-0897,

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"Containment Emergency Sump Performance," dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump performance requirements upon PWRs or boiling-water reactors (BWRs) that were then licensed or under construction, the staff's technical findings identified certain conditions that would inherently lead to these plants' design assumption of 50 percent sump blockage being nonconservative. Therefore, in GL 85-22 the NRC staff recommended that all reactor licensees replace the 50 percent blockage assumption with a comprehensive mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," to reflect the USI A-43 technical findings documented in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events challenged the staff's conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barsebäck Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, ECCS strainers twice became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the "A" loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994; Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995; and Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). The NRC staff has concluded that all BWR licensees have adequately addressed these bulletins.

However, the findings from research to resolve the BWR strainer plugging issue in the late 1990s raised questions concerning the adequacy of PWR sump designs by confirming what the aforementioned BWR strainer plugging events had earlier indicated: (1) that the amount of debris generated by a HELB could be greater than estimated by the USI A-43 research program, (2) that the debris could be finer (and, thus, more easily transportable), and (3) that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These BWR research findings, which may also affect the performance of PWR sumps, prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The objective of GSI-191 is to ensure that post-accident debris blockage would not impede or prevent the operation of the ECCS and CSS in the recirculation mode at PWRs in the event of a LOCA or other HELB accidents for which sump recirculation is required.

Discussion

In the event of a HELB within the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, causing damage and generating debris. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. Through transport methods such as entrainment in the steam/water flows issuing from the break and in containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. If the ECCS or CSS pumps subsequently took suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen. The accumulation of this suspended debris on the sump screen could create a roughly uniform mat over the entire screen surface, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris accumulated, the debris bed would reach a critical thickness at which the head loss across it would exceed the net positive suction head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in the recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure.

To assess the likelihood of the ECCS and CSS pumps at domestic PWRs experiencing a debris-induced loss of NPSH margin during sump recirculation, the NRC sponsored a GSI-191 research program, which culminated in a parametric study. The parametric study mechanistically treated phenomena associated with debris blockage using analytical models of domestic PWRs that were generated with a combination of generic and plant-specific data. As documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002, the GSI-191 parametric study concludes that recirculation sump clogging is a credible concern for the population of domestic PWRs. However, as a result of limitations with respect to plant-specific data and other modeling uncertainties, the parametric study does not definitively identify whether or not particular PWR plants are vulnerable to sump clogging when phenomena associated with debris blockage are modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analysis documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports generated by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. The following pertinent technical reports, which cover debris generation, transport, accumulation, and head loss, are incorporated by reference into the GSI-191 parametric study:

- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.

- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.

- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," dated August 2002.

- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.

In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI-191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in the recirculation mode.

First, as a result of the 50 percent blockage assumption, PWR sump screens were typically designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the substantial structural loadings that would occur due to mechanistically determined debris beds that may cover essentially the entire screen surface. Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging and failure. The ECCS strainer plugging and deformation events that occurred at Perry Unit 1— further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993, and Licensee Event Report (LER) 50-440/93-011, "Excessive Strainer Differential Pressure Across the RHR [Residual Heat Removal] Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation," submitted May 19, 1993—demonstrate the credibility of this concern for screens and strainers that have not been designed with adequate reinforcement.

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include "chokepoints," where the flowpath becomes so constricted that it could become blocked with debris following a HELB. For example, chokepoints may include drains for pools, cavities, or isolated containment compartments, and other constricted drainage paths between physically separated containment elevations. As a result of debris blockage at certain chokepoints, substantial amounts of water required for

adequate recirculation could be held up or diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would function successfully. A reduced available NPSH directly impacts sump screen design because the NPSH margin of the ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. The NRC's GSI-191 research identified the holdup or diversion of recirculation sump inventory as an important and potentially credible concern, and a number of LERs associated with this concern have also been generated, which further confirms both its credibility and potential significance. These LERs include:

- LER 50-369/90-012, "Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action," McGuire Unit 1, submitted August 30, 1990.
- LER 50-266/97-006, "Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation," Point Beach Unit 1, submitted February 19, 1997.
- LER 50-455/97-001, "Unit 2 Containment Drain System Clogged Due to Debris," Byron Unit 2, submitted April 17, 1997.
- LER 50-269/97-010, "Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis," Oconee Unit 1, submitted January 8, 1998.
- LER 50-315/98-017, "Debris Recovered from Ice Condenser Represents Unanalyzed Condition," D.C. Cook Unit 1, submitted July 1, 1998.

Third, debris blockage at flow restrictions within the ECCS and CSS recirculation flowpaths downstream of the sump screen is of potential concern for PWRs. For this mode of debris blockage to occur, pieces of debris would need to have spatial dimensions that would allow them to pass through the sump screen's intended openings, or through screen defects such as gaps or breaches, and then become lodged at downstream flow restrictions such as pump internals, high-pressure safety injection (HPSI) throttle valves, fuel assembly inlet debris screens, or containment spray nozzles. In particular, conditions conducive to downstream debris blockage may be present at PWRs with adverse screen defects, and at PWRs where the maximum dimension of the sump screen's intended openings (e.g., the diagonal dimension of a rectangular mesh) is not the most restrictive point in the ECCS and CSS recirculation flowpaths. Downstream debris blockage at restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, downstream debris blockage at restrictions in the CSS flowpath could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Numerous operational events concerning the discovery of inadequate sump screen configurations that could have led to downstream blockage are cited in Attachment 2 to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated July 14, 1998.

Three emergent items have increased the urgency of the NRC staff's efforts to ensure that PWR licensees are aware of and appropriately responding to the above concerns regarding the potential for debris blockage to impede or prevent the operation of the ECCS and CSS in the recirculation mode: (1) an LER submitted by the licensee for Davis-Besse Unit 1 that declared the recirculation sump inoperable, (2) a subsequent LER submitted by the Davis-Besse licensee that declared the high-pressure injection (HPI) pumps inoperable, and (3) an NRC-sponsored risk study concerning operator actions to mitigate sump clogging.

On December 11, 2002, the licensee for Davis-Besse Unit 1 submitted LER 50-346/02-005-01, "Potential Clogging of the Emergency Sump Due to Debris in Containment." In this LER, the licensee stated that the recirculation sump had been declared inoperable as a result of the potential for sump clogging due to unqualified coatings and other potential sources of post-accident debris (e.g., fibrous insulation and improperly applied qualified coatings) and the potential for downstream debris blockage to occur due to a 6-inch by 3/4-inch gap discovered in the screen. The information provided in this LER, and in a public meeting with the licensee on November 26, 2002, also showed that key information documented in NUREG/CR-6762, Vol. 2, "GSI-191 Technical Assessment: Summary and Analysis of U.S. Pressurized Water Reactor Industry Survey Responses and Responses to GL 97-04," dated August 2002, and other assumptions used in the parametric study to model Davis-Besse Unit 1 were not conservative with respect to design basis assumptions regarding sump screen surface area, minimum containment pool depth at switchover to recirculation, and particulate debris generation.

On May 5, 2003, the Davis-Besse licensee submitted LER 50-346/03-002-00, which stated that the HPI pumps had been declared inoperable as a result of the potential for debris to damage the pump internals during the recirculation phase of certain postulated LOCAs when the HPI pumps are required to take suction from the containment recirculation sump. This LER stated that, when an HPI pump takes suction from the recirculation sump, small particles of debris may result in localized erosion of the mating surfaces around rotating parts, and that the flow of sump water that lubricates the hydrostatic bearing (which is drawn from the volute of the HPI pump) could be blocked by entrained debris, resulting in bearing damage.

In February 2003, Los Alamos National Laboratory published the NRC-sponsored technical report LA-UR-02-7562, "The Impact of Recovery From Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency." The report analyzes the potential risk benefit of operator actions to recover from sump clogging events using a generic probabilistic model to demonstrate that the potential increase in risk due to sump clogging could be reduced by approximately one order of magnitude if PWR licensees have appropriate mitigative measures in place.

In response to these emergent items associated with the potential post-accident debris blockage concerns identified in this bulletin, the NRC is requesting that individual PWR licensees submit information on an expedited basis to document that they have either (1) analyzed the ECCS and CSS recirculation functions with respect to the identified post-accident debris blockage effects, taking into account the recent research findings described in the Discussion section, and determined that compliance exists with all applicable regulatory requirements, or (2) implemented appropriate interim compensatory measures to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions while evaluations to determine compliance proceed. The NRC staff

recognizes that it may be necessary for addressees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in this bulletin, and the staff is preparing a generic letter that would request this information.

To assist in determining whether the ECCS and CSS recirculation functions are in compliance with existing applicable regulatory requirements, addressees may use the guidance in Draft Regulatory Guide 1107 (DG-1107), "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated February 2003. The NRC has also published a technical report entitled NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," dated February 2003, which is designed to serve as a reference for plant-specific analyses with regard to whether a sump would perform its function without preventing the operation of the ECCS and CSS pumps. In addition, the NRC staff supports the development of generic industry guidance and the coordination of addressees' responses to this bulletin as a means of increasing efficiency and streamlining the regulatory verification process. Individual addressees may also develop alternative approaches to those mentioned in this paragraph for determining the status of their regulatory compliance; however, additional staff review may be required to assess the adequacy of alternative approaches.

Conditions at specific PWRs are expected to vary with respect to susceptibility to post-accident debris blockage and various options may be available to addressees for preventing or mitigating the effects of debris blockage. For these reasons, addressees that are unable to confirm compliance with all existing regulatory requirements within 60 days in light of the potential debris blockage effects identified in this bulletin may consider a range of possible interim compensatory measures and may elect to implement those which they deem appropriate, based upon the specific conditions associated with their plants. As stated above, the risk benefit of certain interim compensatory measures is demonstrated by the NRC-sponsored technical report LA-UR-02-7562. Possible interim compensatory measures may include, but are not limited to, the following:

- operator training on indications of and responses to sump clogging
- procedural modifications, if appropriate, that would delay the switchover to containment sump recirculation (e.g., shutting down redundant pumps that are not necessary to provide required flows to cool the containment and reactor core, and operating the CSS intermittently)
- ensuring that alternative water sources are available to refill the RWST or to otherwise provide inventory to inject into the reactor core and spray into the containment atmosphere
- more aggressive containment cleaning and increased foreign material controls
- ensuring containment drainage paths are unblocked
- ensuring sump screens are free of adverse gaps and breaches

In addition to the measures listed above, addressees may also consider implementing unique or plant-specific compensatory measures, as applicable. Commensurate with the potential risk-significance of post-accident debris blockage effects, addressees electing to implement interim compensatory measures in response to this bulletin should ensure the interim measures are implemented as soon as practical. The NRC staff recognizes that the implementation of certain compensatory measures involving containment entry may not be feasible until the next outage.

Approximately two weeks after the issuance of this bulletin, the NRC plans to hold a public meeting to further clarify the intent of the bulletin and respond to any questions from addressees regarding the bulletin. The NRC plans to publish the notice for this public meeting promptly after the bulletin is issued.

Applicable Regulatory Requirements

NRC regulations in Title 10 of the *Code of Federal Regulations*, Section 50.46, (10 CFR 50.46) require that the ECCS satisfy five criteria, one of which is to provide the capability for long-term cooling of the reactor core. As set forth in 10 CFR 50.46(b)(5), the ECCS must have the capability to remove decay heat so that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. For PWRs licensed to the General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50, GDC 35 specifies additional ECCS requirements.

Similarly, for PWRs licensed to the GDCs in Appendix A to 10 CFR Part 50, GDC 38 provides requirements for containment heat removal systems, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDCs may credit a CSS to satisfy similar plant-specific licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

Technical specifications pertain to the ECCS and CSS insofar as they require the operability of these systems for the mitigation of certain design basis accidents. Other plant-specific licensing commitments concerning the ECCS and CSS are also documented in the Final Safety Analysis Report (FSAR).

Applicable Regulatory Guidance

Draft Regulatory Guide 1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated February 2003.

Requested Information

All addressees are requested to provide a response within 60 days of the date of this bulletin that contains either the information requested in Option 1 or Option 2:

Option 1: State that the ECCS and CSS recirculation functions have been analyzed with respect to the potentially adverse post-accident debris blockage effects identified

in this bulletin, taking into account the recent research findings described in the Discussion section, and are in compliance with all existing applicable regulatory requirements.

Option 2: Describe any interim compensatory measures that have been implemented or that will be implemented to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions until an evaluation to determine compliance is complete. If any of the interim compensatory measures listed in the Discussion section will not be implemented, provide a justification. Additionally, for any planned interim measures that will not be in place prior to your response to this bulletin, submit an implementation schedule and provide the basis for concluding that their implementation is not practical until a later date.

Required Response

In accordance with 10 CFR 50.54(f), the NRC requires each addressee to respond as described above. The NRC needs this information to verify addressees' compliance with NRC regulations and to ensure that any interim risks associated with post-accident debris blockage are minimized while evaluations to determine compliance proceed.

Within 60 days of the date of this bulletin, each addressee is required to submit a written response that includes the information requested above in the Requested Information section. Addressees who choose not to submit the requested information must describe in their responses any alternative course of action that they propose to take, including the basis for the acceptability of the proposed alternative course of action.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). A copy of the response should be sent to the appropriate regional administrator.

The NRC staff will review the responses to this bulletin and, if concerns are identified, will notify affected addressees. The staff may also conduct inspections to determine addressees' effectiveness in addressing this bulletin.

Reasons for Information Request

As discussed above, recent research and analysis suggests that (1) most PWR licensees' current safety analyses do not adequately address the potential for the failure of the ECCS and CSS recirculation functions as a result of debris blockage, and (2) the ECCS and CSS recirculation functions at a significant number of operating PWRs could become degraded as a result of the potential effects of debris blockage identified in this bulletin. An ECCS that is incapable of providing long-term reactor core cooling through recirculation operation would be in violation of 10 CFR 50.46. A CSS that is incapable of functioning in the recirculation mode may not comply with GDCs 38 and 41, or other plant-specific licensing requirements or safety

analyses. Furthermore, to address the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions, addressees that are unable to confirm regulatory compliance may find it appropriate to implement compensatory measures until a determination can be made. Therefore, the NRC needs the information requested in this bulletin to assess plant-specific compliance with NRC regulations and to ensure the safe operation of PWR facilities as addressees resolve the concerns identified in this bulletin.

Related Generic Communications

- Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.
- Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in the Suppression Pool Cooling Mode," October 17, 1995.
- Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," May 11, 1993.
- Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," February 18, 1994.
- Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," July 14, 1998.
- Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.
- Generic Letter 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.
- Information Notice 97-13, "Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants," March 24, 1997.
- Information Notice 96-59, "Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris," October 30, 1996.
- Information Notice 96-55, "Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions," October 22, 1996.
- Information Notice 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," May 1, 1996.
- Information Notice 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances," February 13, 1996.

- Information Notice 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," October 4, 1995.
- Information Notice 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," November 30, 1995.
- Information Notice 95-06, "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," January 25, 1995.
- Information Notice 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994.
- Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993.
- Information Notice 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993.
- Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992.
- Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," September 30, 1992.
- Information Notice 89-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," December 1, 1989.
- Information Notice 89-79, Supplement 1, "Degraded Coatings and Corrosion of Steel Containment Vessels," June 29, 1990.
- Information Notice 89-77, "Debris in Containment Emergency Sumps and Incorrect Screen Configurations," November 21, 1989.
- Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," May 19, 1988.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this bulletin transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements. Specifically, the requested information will enable the NRC staff to determine whether PWR licensees are in compliance with plant-specific regulatory requirements concerning the ECCS and CSS recirculation functions and ensure the safe operation of their facilities as they resolve the concerns identified in this bulletin. No backfit is either intended or approved by the issuance of this bulletin and, therefore, the staff has not provided a backfit analysis.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this bulletin is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

Federal Register Notification

A notice of opportunity for public comment on this bulletin was not published in the *Federal Register* because the NRC is requesting information from PWR licensees on an expedited basis to assess compliance with existing applicable regulatory requirements and the necessity for interim compensatory measures. As the resolution of this matter progresses, the opportunity for public involvement will be provided. Nevertheless, comments on the information requested and the technical issues addressed by this bulletin may be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555-0001.

Paperwork Reduction Act Statement

This bulletin contains an information collection that is subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). This information collection was approved by the Office of Management and Budget (OMB), clearance number 3150-0012, which expires on July 31, 2003. The burden to the public for this mandatory information collection is estimated to average 150 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or any other aspect of this information collection, including suggestions for reducing the burden, to the Records Management Branch, Mail Stop T-6 E6, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0012), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contacts or lead project manager listed below.

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Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Thermal-Hydraulic Phenomena Subcommittee
Docket Number:	(not applicable)
Location:	Rockville, Maryland
Date:	Friday, August 20, 2003
Work Order No.:	NRC-1044 Pages 1-198

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE

+ + + + +

WEDNESDAY,

AUGUST 20, 2003

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Graham B. Wallis, Chairman, presiding.

COMMITTEE MEMBERS:

GRAHAM B. WALLIS, Chairman

F. PETER FORD, Member

THOMAS S. KRESS, Member

VICTOR H. RANSOM, Member

STEPHEN L. ROSEN, Member

JOHN D. SIEBER, Member

ACRS STAFF PRESENT:

SANJOY BANERJEE, ACRS Consultant

RALPH CARUSO, ACRS Staff, Designated Government Official

SAM DURAISWAMY, Technical Assistant ACRS/ACNW

NRC STAFF PRESENT:

RALPH ARCHITZEL, NRR/DSSA/SDLB

DR. T.Y. CHANG, RES/DET/ERAB

ANTHONY H. HSIA, RES/DET/ERAB

JOHN LEHNING, NRR/DSASA/SPLB

ALSO PRESENT:

JOHN BUTLER, NEI

DR. BRUCE LETELLIER, Los Alamos National Laboratory

I-N-D-E-X

P-R-O-C-E-E-D-I-N-G-S

(8:33 a.m.)

CHAIRMAN WALLIS: The meeting will now come to order. This is a continuation of the meeting of the Thermal Hydraulics Phenomena Subcommittee of the Advisory Committee on Reactor Safeguards which began yesterday. So I don't think I need to read the entire introduction.

I am Graham Wallis, the Chairman of the subcommittee. Subcommittee members in attendance are Tom Kress, Victor Ransom, Jack Sieber, Peter Ford, and Steve Rosen.

Today we are going to consider Regulatory Guide 1.82, Revision 3, entitled "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

This looks like a topic which is significant, at least potentially significant, to safety and poses quite interesting challenges, both technically and from the regulatory point of view. So we're looking forward to your presentation.

I invite Tony Hsia to get us started.

MR. HSIA: Thank you, Chairman Wallis, and members of the committee. My name is Tony Hsia from the Engineering Research Applications Branch in the

Office of Research. With me today on my right is Dr. T.Y. Chang, also in the same branch with me. To his right is Dr. Bruce Letellier, a consultant from Los Alamos National Laboratory.

Also, I see in the audience we have our colleagues from NRR, and this is a pretty extensive effort. As you have seen reading the background information in the Reg. Guide, it can be traced back -- this issue on sump performance -- traced back to even the early '80s. And we spent a lot of time, very extensive effort, in the late '90s until now.

We have worked very closely with our colleagues at NRR. This is a coordinated effort. And just from the outset I would like to state this Reg. Guide 1.82, Revision 3, is applicable to all plant designs, current and future. With the focus -- because it's related to GSI-191, our focus will be on PWR designs.

So if you can -- if you look at page number 2, we have an overview. This basically encapsulates what we're going to discuss today -- background. And we'll go over to the reasons for why we issued Rev. 3 and what Reg. Guides are intended to be used. And also, I'll summarize the activities related to Reg. Guide 1.82, Rev. 3 up to date.

Then, T.Y. will take over and discuss the key revisions in this current Reg. Guide and resolution of public comments. He will select the most significant and most numerous public comments and how we responded to those for your consideration.

After that, the bulk of the discussion will be the summary of the Reg. Guide as well as a discussion of the accident sequences. And we propose to do it in a tag-team approach. T.Y. will focus on the Reg. Guide itself and what the Reg. Guide says, and Dr. Letellier will get into the technical details. And then, T.Y. will wrap it up regarding the research future activities.

Next viewgraph, please.

Just a quick summary of where we have been. Back in 1974, Rev. 0 of Reg. Guide 1.82 was available, and in that Reg. Guide we discussed net positive suction head calculation based on a very simple assumption of 50 percent of the screen was blocked to figure out the NPSH performance.

And then, USI A-43 was started in January of '79. That focused on the containment emergency sump performance. And additional research was performed until 1985; we have issued a Rev. 1 of the Reg. Guide, which is a guidance based on the

resolution of USI A-43, to instead of using a 50 percent blockage, we're going to say that's not sufficient. We're going to have 100 percent blockages, most conservative assumption.

Starting in the '80s, or early '90s I should say, several nuclear plants started from the Barseback plant in Sweden, and several domestic plants -- mostly BWRs -- ran into the sump -- or I should say strainer, suction strainer blockage events. And that really brought a lot of attention to the agency as well as the industry.

Some additional research was done, and in May of 1996 issued Rev. 2. In that, the effort was focused on the revised guidance for the BWRs. And also, NRC issued both in 96-03. That's on potential plugging of the suction strainer in BWRs, and requested licensees to implement measures to ensure that ECCS functions will perform as designed following a LOCA.

Then, in the late '90s -- well, in the meantime, additional research was performed, and we've switched attention more from BWRs to the PWRs, to see how the PWRs would perform. In late '90s, I believe it was '96/'97 timeframe, GSI-191 was issued. That focused on sump performance of PWRs.

And that's where we are today. Basically, at that stage, the Rev. 2 stage, we are asking the industry to assume 100 percent blockage unless they can justify through test or analysis that they can have a more realistic estimate.

CHAIRMAN WALLIS: Well, 100 percent blockage, does that mean that the pumps just cannot pump any water?

MR. HSIA: Assume that 100 percent blockage of the screens.

CHAIRMAN WALLIS: That means the pumps cannot pump any water?

MR. HSIA: No.

CHAIRMAN WALLIS: So we have to assume the pumps are inoperable?

MR. ARCHITZEL: That's one of the recommendations at Rev. 2. This is Ralph Architzel. It just means that you're not having 50 percent -- an arbitrary 50 percent assumption. It's a mechanistic assumption that you had blockage and it can be uniform over the surface. You still get water through. It's an analysis done to say that you --

CHAIRMAN WALLIS: And you have to show -- you assume 100 percent blockage. That means that there is something over the whole surface.

MR. ARCHITZEL: A hundred percent coverage of the surface.

CHAIRMAN WALLIS: In order to figure out whether the pumps will work or not, you have to know what that stuff is.

MR. ARCHITZEL: Exactly.

CHAIRMAN WALLIS: And so you haven't really, with this assumption, given enough information to solve the problem.

MR. HSIA: Correct. That's why we're continuing to do research, and that's the most conservative way to do it at that time.

CHAIRMAN WALLIS: Well, it isn't really conservative yet, because you haven't said what the blockage consists of. It could be 100 percent of insignificant stuff.

MR. HSIA: Could be.

CHAIRMAN WALLIS: So it's not really conservative yet.

MR. HSIA: Okay.

CHAIRMAN WALLIS: Until you say what that stuff is.

MR. HSIA: Correct.

CHAIRMAN WALLIS: You said it was blocked so much that the pumps couldn't work. That seems to

me is a conservative assumption. Otherwise, it doesn't say anything. It just says there is something on the screen everywhere, and that doesn't really say anything until you say what you mean.

DR. LETELLIER: I think you'll see in the research efforts that the debris generation and transport tests have, in fact, characterized what types and amounts of material might arrive on --

CHAIRMAN WALLIS: Might, yes. Might. But I read in your report there are 13,000 cubic feet of fiber in some of these -- in the air handling equipment, for instance, in the containment. Now, if any one percent of that gets on a screen, it blocks it completely.

DR. LETELLIER: There's the potential for 100 percent coverage with an attendant head loss associated with that bed.

CHAIRMAN WALLIS: Yes, it's all potential. It's all "it might happen." It's not an assumption that lets you calculate anything yet.

MR. HSIA: Correct. That's why we at this point have stuck -- continued to gain knowledge. Right now, at this stage, our thought was the plant needed to do plant-specific analysis. Some plants may not have that kind of issue. It depends on how -- the

probability of their break and where they assume the break.

So that brings us to where we are today is Rev. 3. And we're here today -- hopefully, we'll -- our plan is to have Rev. 3 -- with your approval, we'll issue the Rev. 3 shortly.

CHAIRMAN WALLIS: Oh. We can stop it?

MR. HSIA: Correct. Correct.

CHAIRMAN WALLIS: Thank you.

MR. HSIA: Hopefully, that's not the outcome we're here for. The reason --

CHAIRMAN WALLIS: It depends on how much blockage we want to insert in your process.

MR. HSIA: Correct. And we have to find ways to justify it.

CHAIRMAN WALLIS: Okay. Yes, please. Please do that.

MR. HSIA: Okay. The reason for issuing Rev. 3 is to contribute to the resolution of GSI-191, to enhance the blockage evaluation guidance for PWR, and to provide guidance to make sure we put out there methods acceptable to the staff, because, like I said earlier, Rev. 2 -- we felt that Rev. 2 of Reg. Guide 1.82 was not comprehensive enough to ensure adequate evaluation of PWRs susceptibility.

I just want to clarify that Reg. Guides are not a substitute for regulations, and compliance is not required. And we will talk a little bit about alternative methods that -- as a matter of fact, that's one of the --

CHAIRMAN WALLIS: That's rather funny. You know, they have this thing that compliance is not required. So that's why I couldn't understand about this whole exercise. Out goes this Reg. Guide, and it looks like a really serious matter. And it's quite likely, it seems, that some -- quite a few plants will not be able to meet all of these requirements in this Guide. So what then happens?

MR. HSIA: Okay. Yes. The Reg. Guide points out one or several acceptable methods in --

CHAIRMAN WALLIS: But it's also a subpart of the regulations. So what happens when they can't do it?

MR. HSIA: If they cannot do it, or they choose not to use the methods described in here, they can come up with their own methods. And that's the time that they have to send it in here. Either way, they have to send it here for --

MEMBER KRESS: It's in the regulations that they have to assure that you can do the longer-

term cooling and --

MR. HSIA: Yes. I was about to say --

MEMBER KRESS: There's a requirement there.

MR. HSIA: -- there are requirements in long-term cooling, 50.46.

MEMBER KRESS: So it's not like --

MR. HSIA: That's a regulation that they have to satisfy. But they don't have to use the method described in --

CHAIRMAN WALLIS: But they have to use some method.

MR. HSIA: Yes. So like I said, when they chose or they cannot use this Reg. Guide methods, they can come up with their own through experiments, through tests, and then we need to evaluate -- assess that.

MEMBER KRESS: They have to satisfy you guys that --

MR. HSIA: Absolutely.

MEMBER ROSEN: And then would you come talk to us about that, if that unlikely event occurred, someone chose to do it their own way?

MR. HSIA: I would like to ask one of my colleagues at NRR if that's -- that's a regulatory

issue. If they come in, you are going to issue an SER. Do you come in front of ACRS? I don't -- I'm not sure they come to you for every plant they come in for -- with different methods.

MEMBER ROSEN: Well, it seems reasonable to me that if you're asking for us to agree that this general method should be applicable to everybody -- and we do --

MR. HSIA: Right.

MEMBER ROSEN: -- and somebody else chooses another method, you ought to come in and ask us whether or not the other method is --

MR. HSIA: Well, one scenario could be if that method, when they reviewed the alternative methods, they will still check based on this method to see if they are compatible, if they're similar. And if they find there are large discrepancies, they believe they may choose to come in front of the ACRS.

But if they conclude it's a different method but it's very similar, and it's technically sound, they may not come in front of ACRS.

MR. ARCHITZEL: I guess just for going forward, and a future plant would come in using this, we would review it like Tony is saying. And you'd review when the SER came forward for that plant in the

ACRS. And if an issue arose to that, you'd hear about it, you know, or you'd see it in the SER.

For the existing plants, the backfit comes to play -- and not necessarily all of the positions in the Reg. Guide would be imposed on the plants that are out there. There are selected positions that would be imposed.

CHAIRMAN WALLIS: So you're saying backfit comes to play, because at the end of this Guide it says, "No backfitting is intended or approved," or something.

MR. ARCHITZEL: That's right. So as we go forward, we're not allowed to backfit provisions in this Reg. Guide without going through --

CHAIRMAN WALLIS: Well, it just seems to me that --

MR. ARCHITZEL: But we will use it --

CHAIRMAN WALLIS: -- it may well be that backfitting will have to occur as a result of studying this issue.

MR. ARCHITZEL: But the current plan is to ask if -- for the current plants, ask them for information. And that's not exactly a backfit. They have to do an evaluation. So it's the way the generic communication process works.

CHAIRMAN WALLIS: It's putting off today when they have to do something, it seems to me.

MEMBER ROSEN: Yes. But it may not be a backfit. It may be -- because they have to provide long-term cooling.

CHAIRMAN WALLIS: Yes.

MEMBER ROSEN: And that's the field upon which the agency issued a license.

CHAIRMAN WALLIS: Yes.

MEMBER ROSEN: And if it's now found that the long-term cooling is threatened, or not likely, then it's not a backfit for them to fix it, so that they restore long-term cooling.

MEMBER KRESS: It's a backfit, but they don't have to do a regulatory analysis.

CHAIRMAN WALLIS: It just bothered me to state that no backfitting is intended. It may well be that backfit is the right thing to do, so it's -- dismissing backfit out of hand is not -- didn't seem to me appropriate. Perhaps we'll get to that later on.

MR. HSIA: Okay.

MEMBER ROSEN: You don't need a cost-benefit analysis, a 51-09 analysis, to --

CHAIRMAN WALLIS: But if you don't meet

the regulations for long-term cooling --

MR. HSIA: If you don't meet the regulations, that becomes a compliance issue.

CHAIRMAN WALLIS: Right. You must do it.

MR. HSIA: Right.

CHAIRMAN WALLIS: Okay. So perhaps we should go on, then, from there.

MR. HSIA: Okay. Viewgraph 5 is a brief history. We were here, briefed the subcommittee back in February '03. As you can see, several of the actors have changed. I wasn't here at the time and neither was T.Y. As a matter of fact, our able staff member is now working for -- for you now, ACRS staff. So, but T.Y. is as competent as B.P., so I'm very pleased.

So at that time, we were here, and so was NRR. They discussed GSI-191 and the plans for -- they have issued a bulletin since then, and they are planning to issue a generic letter early next year.

The draft Reg. Guide at that time was called DG-1107 -- was issued for public comment from February to April. And we have -- T.Y. will discuss the resolution of those comments.

CHAIRMAN WALLIS: I'd like to ask you about resolving public comment. We're going to get to

this in detail. I read the Guide, and I had almost all of the same comments that the commenters had, even though you've already addressed them, you say.

So it's a bit of a puzzle. Are public comments resolved simply by you saying, "We've resolved them," or do you have to go back to the public and show that you have answered the question? I mean, are you like the politician who gets one question and answers it with something else, or answers it with something which doesn't really answer it? What's the assurance that this resolution really answers the comments in an effective way?

MR. HSIA: Are you saying, how do we get back to the comment --

CHAIRMAN WALLIS: No. You say you have resolved public comments. I mean, are you the arbiter of whether or not you have answered the comments effectively?

MR. HSIA: In a way, yes, we are. We are doing the best we can to say, "This is how we plan -- how we propose to resolve the comments." That's why we're here today.

CHAIRMAN WALLIS: Okay. Check on whether or not you have done this right. It's your own professional --

MR. HSIA: Correct.

CHAIRMAN WALLIS: -- integrity and values, and so on, or maybe the ACRS.

MR. HSIA: Exactly. That's why we're here today as well as this is a public meting. If any public here wants to say, "Hey, you didn't answer my question" or "I don't agree" --

CHAIRMAN WALLIS: Okay. That's so they could come back.

MR. HSIA: Yes.

CHAIRMAN WALLIS: Okay.

MR. HSIA: Yes.

MEMBER KRESS: That's the way you always did it.

MR. HSIA: Yes, correct.

MR. BANERJEE: May I just make a comment here, Mr. Chairman. As I understand, the process is when a rule is proposed or a Reg. Guide is proposed, you send out for public comments. When the comments are received, the staff members analyze the comments. And then, when you finalize any document, it goes back out again with your detailed analysis of each comment and the response that the staff is proposing.

And if there is any serious problem then, then the public comes back, and either in the form of

a petition or in the form of a letter to the Commission -- so then the -- the process is very clearly marked, and it's a cycle.

So the way I understand right now, the staff is coming in front of the subcommittee here to tell their plan to resolve the public comment. If you have any serious doubts or anything, then the staff will go back and then make corrections before they go out for their final product.

Isn't that correct, Tony?

MR. HSIA: That is correct. And you can see from the fourth bullet on this viewgraph we're here today, and we are -- to make sure we're not -- because we say this is not a backfit. That's why CRGR has -- we'll have a meeting with them later on this month.

And as you can see, we are coming back; you have another shot at us. I don't mean literally, but --

(Laughter.)

CHAIRMAN WALLIS: You want a letter to -- it seems to me you want a letter in September.

MR. HSIA: After September 11th, yes.

CHAIRMAN WALLIS: That's where if we have still comments, or we think you haven't --

MR. HSIA: Yes.

CHAIRMAN WALLIS: -- the comments, we say so, and then you have another shot at resolving them, right?

MR. HSIA: Correct. We'll make another attempt, I would say.

CHAIRMAN WALLIS: But you're going to issue a 903 anyway?

MR. HSIA: We would like to. But obviously, if there's issues we cannot resolve, that's not going to happen.

CHAIRMAN WALLIS: Okay.

MR. HSIA: And that ends my part of the presentation. I would like to turn it over to T.Y.

DR. CHANG: My name is T.Y. Chang, Office of Research.

This slide shows that once the majority of the revisions are made, it's made in the PWR sections, because this is the intention of issuing this Reg. Guide. However, we tried to make sure that the PWR sections and the BWR sections are consistent with each other whenever it's appropriate.

Another thing is Reg. Guide 1.1 has been subsumed into this current version. So only for some older plants they have to refer back to this Reg.

Guide 1.1. For future plants, they refer to Reg. Guide 1.82 now for the NPSH issue.

Next slide, please.

This is a summary of the public comments we received. We received 89 comments from seven commenters -- four utilities, Westinghouse, NEI, and one individual.

And the last bullet, in descending order, are frequencies of comments raised. We have -- the first one we received 13 comments, and the second one eight comments, and so forth. We are going to go each one now.

Next slide, please. Yes?

MEMBER KRESS: Just a general thought. It seems to me every time we review some of these draft Reg. Guides and rules, and you guys go out for comments and then get them back, 99 percent of the comments come from industry -- utilities, Westinghouse, NEI. Once in a while we get one from the Union of Concerned Scientists, and sometimes an individual.

But is that an appropriate -- you know, all we're doing is talking to the utilities, it seems like. How do you distribute? Do you just put in the Federal Register Notice and then --

DR. CHANG: Yes, it's announced in the Federal Register Notice. Anyone can send in their comments.

MEMBER KRESS: Anybody can that wants to.

DR. CHANG: Right.

MEMBER KRESS: This individual, is that a public citizen, or did it -- or do they belong to some organization?

DR. CHANG: I think he's a consultant.

MEMBER KRESS: Consultant.

DR. CHANG: Yes.

MEMBER KRESS: It always bothers me that we don't seem to get real public input to these things. We seem to always be -- hear from the industry only.

MEMBER ROSEN: People don't vote either.

MEMBER KRESS: Yes, that's true.

MEMBER SIEBER: Well, people don't get a subscription to the Federal Register. You know, you get about this much stuff, two feet high, every day, because there's a lot of agencies, a lot of stuff in there.

MEMBER KRESS: I just wondered if there was a better way to do it, but I can't think of one.

CHAIRMAN WALLIS: Well, I would -- yes, I

would think not so much the public, but sort of a technical savvy community.

MEMBER KRESS: Yes.

CHAIRMAN WALLIS: So if somebody who is not part of the system of regulation and licensing, and all of that, were to read this, would it seem believable? Trying to get some view which isn't -- doesn't have a motive, profit motive or something. Are we the only people like that?

MEMBER KRESS: I don't know.

MEMBER RANSOM: Well, the Union of Concerned Scientists, usually they have a motive, too.

MEMBER KRESS: No. They have an agenda. Sometimes you can believe them; sometimes you can't.

DR. CHANG: Okay. Let me go on.

CHAIRMAN WALLIS: Well, we've made this comment many times. I think it is a weakness in the system. These comments always come back from interested parties trying to do something for their own benefit.

DR. CHANG: I think that's human nature, right?

Next slide. This --

CHAIRMAN WALLIS: I don't know what benefit I'm getting out of being here.

(Laughter.)

MEMBER KRESS: What was the comment -- are you going to go over these comments later?

CHAIRMAN WALLIS: Yes.

MEMBER KRESS: Okay.

CHAIRMAN WALLIS: Let's go on.

DR. CHANG: The next one is about conformance issue for current plans. We've got 13 of them.

CHAIRMAN WALLIS: I think this is an important issue.

DR. CHANG: Yes. For instance, the first comment is how Reg. Guide will be used for the current plans. We mentioned that there is no intention for backfitting for the current plans. It's only used as simply for the evaluation of the long-term cooling of the ECCS.

CHAIRMAN WALLIS: No, I'm not sure you can use the Reg. Guide for evaluation methodologies, because my impression is the Reg. Guide says, "Go and do this. Go and do that. Go and see" --

DR. CHANG: Well, we have --

CHAIRMAN WALLIS: It doesn't say anything about the existence of a methodology for doing it.

DR. CHANG: We have staff positions there,

too. Okay? Not only acceptable methods.

CHAIRMAN WALLIS: It all seems so vague. It says, "Go and calculate the debris transport." It's not clear that anybody knows how to calculate the debris transport.

DR. CHANG: Well, this is not the intention of the Reg. Guide. We have not tried to be prescriptive that people have to follow those steps.

CHAIRMAN WALLIS: Do you see the problem I have? You're evaluating methodologies which probably don't exist.

MR. HSIA: If I may jump in. Bruce, please. Welcome. Go ahead, Bruce.

DR. LETELLIER: Well, first of all, I don't think the Reg. Guide can be applied without a knowledge of the historical research base that goes along with it. And there has been an attempt to document the supporting references. And one suggestion has been that we add citations in the appropriate sections, so that it's not difficult to reconstruct that history for a first-time user.

MEMBER KRESS: That would seem to be helpful to the reader.

DR. CHANG: Yes. I think that that's the intent of the second part of my presentation is to try

to describe our staff positions in the Reg. Guide and the so-called acceptable methods. And then, Bruce will go into specific ways of how to apply those methods in real cases.

So I think that will address your question. Just be patient with us.

CHAIRMAN WALLIS: Well, I don't think it helps at all. If you read the Guide, you just pick up at random a section, all insulation, blah, blah, blah, blah, blah, blah, blah, you know, great list of stuff, should be considered the debris source. Models or experiments should be used to predict the size of the postulated debris.

DR. CHANG: That's one of the acceptable methods. They can choose to be conservative, to assume the worst --

CHAIRMAN WALLIS: But all postulation is an enormous amount of stuff.

MEMBER KRESS: Well, they give some guidance in the Los Alamos report on how to deal with that.

MR. HSIA: Chairman Wallis?

MEMBER KRESS: I think if you reference the Los Alamos methodology in there, it might help.

MR. HSIA: Chairman Wallis, this is Tony

Hsia from Research.

CHAIRMAN WALLIS: Yes.

MR. HSIA: This Reg. Guide is not meant to be a manual for anybody who wants to assess their plant vulnerability regarding debris impact on ECCS performance. It is, indeed, a guide. In there later on I hope you will see that we -- like you just read, we have guidance here saying, "You shall do this. You should do that. We recommend that."

And many, many of those, if not all, have been documented based -- as a result of previous research and numerous reports, NUREG reports. I just want to mention two very significant ones. One is a knowledge-based report. I'm sure you all have a copy. Another one is an older report, NUREG/CR-6244. Both of those have been peer reviewed, and so this is nothing new, really, to the industry or anybody.

CHAIRMAN WALLIS: Well, see, the problem is I don't know how you got this -- this knowledge base. But I read it, and it's so qualitative.

MR. HSIA: I'm sorry?

CHAIRMAN WALLIS: It describes things, and it describes things you ought to consider. It doesn't say how to do it. It says, "Here's this event in Barseback. This is what happened. Here's this thing

in Hyse, Dumphrey, Oktor," or so and so," all these things. It describes it. It doesn't give the impression that there's any way to predict what happened.

MR. HSIA: That part, you're correct. That is early in that report. Later on --

CHAIRMAN WALLIS: What good is it for predicting anything?

MR. HSIA: Later on there are sections into different -- each phase of the accident sequences. There are methods described, a test that was done, and what you can learn from those tests, as well as the analyses that was done, what you can do with those analyses/methods. And that's what we're hoping -- that Bruce will get into that detail as we go along.

CHAIRMAN WALLIS: Okay. So we'll get into that detail later.

MR. HSIA: Right. The point I want to stress right now is both of those reports I mentioned earlier have been peer reviewed, and discussions we have had --

CHAIRMAN WALLIS: Your process must be something like the public comment process, too.

MR. HSIA: Well, public comments --

anybody in the public, peer review, our field industry experts, or a professional engineer that has expertise in this area. There are technical people who took time to really review all of the reports.

And also, we have had several workshops. We have had discussions with the public, with the interested parties. So many of those methods, experiments, analyses, have been discussed before. So I just want to say this is not brand-new to the people who are interested in doing this.

CHAIRMAN WALLIS: I just think when you have a peer review you have to have some sort of -- be clear what it is they're reviewing, for what purpose. And a peer review that says, "This is an interesting document" is one thing. A peer review which says, "This document really explains how to make calculations for something with some kind of accuracy" is a really different kind of peer review.

DR. LETELLIER: I think you're expecting to see predictive phenomenology models that simply don't exist.

CHAIRMAN WALLIS: That's right. Well, in order to do what you want done in the Reg. Guide, I have to have those.

DR. LETELLIER: I think the objective of

the Reg. Guide is to make a conservative, yet realistic, approximation of the various stages of the accident sequence.

CHAIRMAN WALLIS: Okay. Maybe you'll make that case. I want to let you get on to it.

DR. LETELLIER: I hope so.

CHAIRMAN WALLIS: Yes. I'm sorry to interrupt you, but you were talking about this conformance issue.

DR. CHANG: Right, the first bullet. And then, the second comment on the conformance issue is some current plans have different designs as compared to the ones we mention in the Guide. For instance, the multiple --

CHAIRMAN WALLIS: Do any of them have floors that slope away from the screens? That seems a strange requirement.

DR. CHANG: Yes. I don't know whether -- are there --

DR. LETELLIER: Not to my knowledge. In fact, there are very -- perhaps one or two at the most that actually have designed drainage systems to return water to the screen.

CHAIRMAN WALLIS: In any shower stall or anything, the drain is at the bottom of the slope, not

on the top of the hill. It seems a very strange statement in there. So they certainly have different designs as compared to the RG position in terms of the slope of the floor.

DR. CHANG: Yes. We tried to say that --

MR. ARCHITZEL: But there are some that have that, and -- but the normal sump would be the lowest point. The accident sumps, there's quite a few that do have the --

CHAIRMAN WALLIS: They do.

MR. ARCHITZEL: -- slight rise.

CHAIRMAN WALLIS: Okay.

MR. ARCHITZEL: Certainly, a lot with curbs.

CHAIRMAN WALLIS: Okay. So they do.

MEMBER ROSEN: This is not an accident, then. The curbs are there for a reason.

CHAIRMAN WALLIS: This means that when there are spills of water it goes on the floor and is not drained because the highest point is --

MR. ARCHITZEL: No, there's a normal sump that would be the lowest point in the drain, different sumps.

CHAIRMAN WALLIS: Oh, okay. Thank you. That's good. That helps.

MEMBER KRESS: Multiple sumps don't seem like they're any different to me than one sump. It's just like a bigger one sump. Is that --

DR. CHANG: Well, you have two independent sumps in different locations. I think usually --

MEMBER KRESS: Yes, but there's a common cause failure, and that's the debris goes to both of them. It's like just having one sump that's a little bigger than this one.

MEMBER SIEBER: If your containment is compartmentalized --

DR. CHANG: Right.

MEMBER SIEBER: -- then you have a different debris field for one --

DR. CHANG: That's far away from each other.

MEMBER SIEBER: -- than you have from the other.

MEMBER ROSEN: You have a longer transport there for the one -- distance to one sump than the other, and that may be important.

MEMBER KRESS: I can see that being important.

DR. CHANG: But our intention is that this Guide is not just for current plans. It's for future

plans as well. So we just pointed out those possibilities for the consideration if future plants are being designed.

The third comment is the Reg. Guide appears to favor a particular configuration of screen because of the cartoons we have in the Reg. Guide. We tried to clarify, to change the caption, saying that those are conceptual features -- to indicate that they are conceptual in nature.

CHAIRMAN WALLIS: So this Reg. Guide will be used in the first response here, the evaluation of current licensees, methodologies, long-term recirculation cooling, and this will be, then, accompanying some NRR effort to make sure that the plants actually have those capabilities.

DR. CHANG: Oh, yes. Oh, yes.

Ralph, do you want to talk about the NRR continuing program on this issue?

MR. ARCHITZEL: Yes. We were here before at the same time you were. We currently plan -- last time we were here we had a Generic Letter in front of you, and you said, "Put it out quickly." We ended up splitting that after we met with you into a bulletin with the interim actions and a Generic Letter to follow.

The Generic Letter will require evaluations and request information to show that they can meet this deterministically. But before -- or what they're going to do that to is not this Reg. Guide. It's going to be -- at the present plan, we plan on looking at industry evaluation guidelines, detailed guidelines, in terms of how you do these evaluations. So that there's more of a --

CHAIRMAN WALLIS: But that isn't available yet, is it?

MR. ARCHITZEL: It's not available yet. The last plan from NEI we heard was September of this year, and that may not make that date.

We would evaluate that and write an SER, and that guidelines it. We're looking towards the middle of next year to complete our evaluation of those guidelines.

CHAIRMAN WALLIS: Well, I guess from the public point of view, the issue is how long it's going to take to resolve what's been a long-standing safety issue of impossible importance.

MR. ARCHITZEL: NEI approved guidelines are a while off yet. But this would be the yardstick we would use to evaluate those guidelines. This Reg. Guide is used as the evaluation of --

CHAIRMAN WALLIS: Okay. So it's a yardstick. So it better be -- better have units on it, right?

MEMBER FORD: Since you don't have a predictive methodology -- I mean, for instance, you cannot predict why Barseback or Gundremmingen, these other stations which have seen pump blockage occur, that a whole list of various variables -- mesh size, debris sources, etcetera, where you have no way of quantifying whether that particular lineup or parameters will give you a real -- give you a problem down the sump.

So following on from the previous question, what is your criteria for success or compliance by the utility to this Reg. Guide? This Reg. Guide just lists a whole lot of, "Hey, look out for the slope of the floor, mesh size," etcetera, etcetera. You're just listing all of the variables, but you're not giving any criteria as to the well, which are the most important ones.

What defines compliance to the Reg. Guide? Do you understand what I'm saying? There's no quantification.

DR. LETELLIER: Well, let me attempt to clarify. First of all, the units, the calibration of

the yardstick, are based on NPSH margin. That is the ultimate condition of compliance -- whether or not a given licensee can accommodate a certain fraction of debris transport and still provide long-term cooling as defined by --

MEMBER FORD: But there is no algorithm relating NPSH to all of these other variables.

DR. LETELLIER: Well, when we say that there are no predictive models, in large part we're referring to the transport step. Now, we do have test data that describes debris generation. We have test data that describe head loss when the debris arrives on the screen. And those are predictive; they're based on empirical correlations and on some semi-empirical theory.

So the various pieces have been quantified to the level of detail that was possible with the resources that we've been given in the past few years. The lack of predictive capability comes in in the variability of input parameters.

We're not certain exactly what the conditions of a given accident will be, and we're -- we don't have a capability to predict the transport fate of an assumed particle of debris.

MEMBER FORD: That's right.

DR. LETELLIER: And so, therefore, we're using the test data to make conservative engineering judgments about the connections between each step of transport.

MEMBER FORD: I guess I'm putting myself in the shoes of the utility, and saying, "Okay. I've got to meet a certain NPSH quantitative criteria." But I have no idea what -- the things I should be controlling. And I've got this great big list of things, and if you look at your report, the Los Alamos report, there's a huge number of interrelations which no one -- no one -- understands or can predict.

DR. LETELLIER: Well, I --

MEMBER FORD: So is there going to be a big EPRI program to put a -- to qualify this, so they can react proactively to this problem?

DR. LETELLIER: I'd prefer to respond to a specific question regarding lack of predictive capability, and that way we could show you the supporting evidence that would help you make judgments.

MEMBER FORD: Okay.

DR. LETELLIER: But, in general, let me say that the guidance is intended to demonstrate acceptable methods that range all the way from

100 percent damage, 100 percent transport, 100 percent blockage, all the way down to phenomenology-based engineering judgments about what fractions would actually participate in each step of the process.

Of course, the more detail that you have to take credit for, then the more responsibility you have to baseline your judgments on data, testing, and evaluation programs.

In fact, when the comment was made by Dr. Wallis about 100 percent inventory being overly conservative, in fact, that was the resolution path taken by the BWRs. As a matter of practicality, they had enough space to redesign their strainers to accommodate that amount of material.

CHAIRMAN WALLIS: Well, it couldn't be all the material in the air handling units.

DR. LETELLIER: They designed their strainers to accommodate all of the insulation, thermal insulation, in containment.

CHAIRMAN WALLIS: Well, 13,000 cubic feet? That's this room full. I don't know, maybe more than this room full. Yes, more than this room full. I can't believe it, that you're going to put all of that on your strainer.

DR. LETELLIER: You're referring to filter

media in the air handling units and --

CHAIRMAN WALLIS: It's in your -- this technical basis. I just -- it just struck me.

DR. LETELLIER: Well, keep in mind --

CHAIRMAN WALLIS: I don't know what this is, and why it's in the air handling units. But it's in your report that it's there.

DR. LETELLIER: Keep in mind that when you consider debris generation, you have to examine the potential source locations. And then you assess the targets that might be impacted by the damage, so --

CHAIRMAN WALLIS: I have no idea where the air handler units are relative to where the LOCA might be or why --

DR. LETELLIER: And it may vary --

CHAIRMAN WALLIS: -- the stuff might fall out in a steam environment or not. But that's something that presumably is going to be calculated using your methods.

DR. LETELLIER: The locations may vary widely.

CHAIRMAN WALLIS: Yes.

DR. LETELLIER: In fact, and it -- I think it's listed there for completeness sake. If a particular licensee knows that their air handling

units are vulnerable to impingement, then that represents a potential debris source that they have to accommodate.

CHAIRMAN WALLIS: I wonder if they have any clue about whether they're vulnerable to a shockwave.

DR. LETELLIER: By inference of proximity, and based on the test data for damage zones for different types of debris ranging from bare fibrous insulation all the way to encased stainless steel jackets, I think that the industry does have a good impression of what the damage zones are.

Now, that's not to say that the database is entirely inclusive. We were able to test the predominant materials, the predominant insulation types. But there are certainly others.

CHAIRMAN WALLIS: Well, that's what's -- I do see you have these -- you have -- I know you have some good tests on certain kinds of insulation on pipes. But this air handler unit, where is it? I'm sorry to keep on this, but because this is a huge number in your report -- 15,000 cubic feet.

Now, is this in sheets of loose stuff in some kind of -- like in my domestic heating system, hot air system? It's a very, very flimsy filter.

DR. LETELLIER: But in general --

CHAIRMAN WALLIS: The slightest thing can break that up.

DR. LETELLIER: That's true. But in general --

CHAIRMAN WALLIS: Is that what they're like?

DR. LETELLIER: Yes.

CHAIRMAN WALLIS: Then, why are they considered?

DR. LETELLIER: But in keeping with your analogy of a home furnace system, you know that those materials, those fiberglass panels, are encased in mechanical equipment. They are shielded, in a sense, by the sheet metal.

CHAIRMAN WALLIS: Well, I don't know how they are in the plant.

DR. LETELLIER: In fact, that's true in the plants as well.

CHAIRMAN WALLIS: So that the utility has to look very carefully at all of those things, like say the air handling units, and say, "Gee whiz, my filters are not very well protected. I'd better do something about it," or something?

DR. LETELLIER: That's true. Ultimately,

the --

CHAIRMAN WALLIS: Do all of these assessments and --

DR. LETELLIER: -- judgment falls on them. But keep in mind --

CHAIRMAN WALLIS: Eventually, maybe if they don't do it, some NRC inspector will walk around, if they have a walkaround in site containment, and say, "Gee whiz, I can see a lot of loose filter material up in that air filter. This looks like something that might give a problem with sump block issue at" --

DR. LETELLIER: Well, the guidance also serves the purpose of audits for the regional inspectors. And so the Reg. Guide provides consistency between the NRC approach and the industry's perspective as well.

Keep in mind that the assessment of a given vulnerability may be as simple as proximity. This is outside the damage zone. Therefore --

CHAIRMAN WALLIS: We're going to move on.

DR. CHANG: Yes.

CHAIRMAN WALLIS: But you're putting an awful lot of reliance here on the ability of each one of these licensees to make a proper assessment of all

the sources of debris and what will happen to it.

MEMBER ROSEN: That's correct. When we get the guidance from NEI --

CHAIRMAN WALLIS: Right. We have to get some guidance. We haven't got it yet.

MEMBER ROSEN: No, we haven't got it.

CHAIRMAN WALLIS: We have no idea if it's going to be adequate.

MEMBER ROSEN: Well, we will presume that they will do a good job as they do on many things and be proud of it and tell us about it.

I would observe it's 9:15. That's the close of the discussion on the comments, and I don't think we're quite there.

DR. CHANG: Okay. The next issue about overpressure -- in the Reg. Guide, we mentioned that for the ECCS and containment heat removal systems, they should be designed such that the pumps have available sufficient to the NPSH.

Assuming no overpressure from -- as compared to that -- before the LOCA -- this is a conservative assumption -- the comment is that this is not consistent with the licensing basis for certain subatmospheric containment plants, because in those plants they have vacuum under the normal operation

condition.

Our response is that the original position stays with change -- with some modifications. The modification is that we said for subatmospheric containments, this guidance should apply after the injection phase has terminated. Prior to termination of the injection phase, the analysis should include conservative predictions of the containment subatmospheric pressure and sump water temperature as a function of time.

MEMBER KRESS: Why should you give subatmospheric containments this advantage but not give it to the other plants? Why shouldn't an ordinary large dry PWR be able to take the containment pressure prior -- after injection also? If it's good for one plant, shouldn't it be good for the other?

DR. CHANG: Well, I think it's consistent. We are trying to be on the conservative side.

MEMBER KRESS: Yes, you've been consistent. The subatmospheric plants have been given --

DR. CHANG: For subatmospheric plants --

MEMBER KRESS: -- an allowance for overpressure.

DR. CHANG: Prior to the switchover, they

have to assume conservative predictions for pressure and water temperature as a function of time.

MEMBER KRESS: Yes. But it seems to me like if you're going to let the subatmospheric containments do that, you ought to let other containment types do it also.

DR. LETELLIER: To be honest, I'm not certain what additional benefit that really adds. If you look at the words that -- we're talking about the switchover to recirculation, between injection and recirculation. And after the injection phase has terminated, the guidance defaults back to the pressure that existed before the --

DR. CHANG: They still have to comply to the pre-LOCA condition.

DR. LETELLIER: In effect, T.Y. was correct that our -- the staff position has not changed, that we're defaulting back to the Reg. Guide 1.1 position, that in order to accommodate a variety of accident scenarios, including loss of containment, it's always conservative to assume the pressure that occurred before the LOCA event.

DR. CHANG: Okay. The next one, the next slide, please, is on the screen mesh size. In the original Reg. Guide sent out for public comment, we

have a sentence saying that a site should be smaller than the minimum restrictions found in the systems downstream of the sump, and then later the ECCS or the reactor coolant system components.

The comment is that this may lead to very high head loss for -- in current screens, if you use such a small mesh. And also, it may make the screen areas too large to be practical.

The second comment on the mesh size is someone suggested that the long thin debris slivers may pass axially through the sump screen, and may then reorient and clog the flow restrictions downstream, such that pump seals -- such as pump seals and barriers in those locations. This shall be considered -- this is the comment.

Our response to the first one is that we modified the Reg. Guide to say that the size of the screen pump opening should be determined considering the flow restrictions of systems. We don't say it has to be smaller.

And then, the mesh size is -- if the mesh size is impractical to be fine enough to filter out particles of debris that may cause damage to the downstream equipment, then it is expected that modification would be made to the ECCS pumps, or they

can purchase a pump that can handle those small particles.

And on the second comment we --

CHAIRMAN WALLIS: There is a pump that will do this that's easily accessible and --

DR. CHANG: Ralph, do you have any information on that?

DR. LETELLIER: We don't have specific vendor information, but we are aware of pumps that are designed to handle high debris loadings. We're not certain that they're qualified for nuclear applications.

The point is, in the response to this comment, is that the filter screens have a performance criteria. They are there for a purpose -- to protect downstream equipment. And the vulnerabilities of the downstream equipment should be used to define the performance standards.

MR. ARCHITZEL: But I guess to go to a specific example, in the Davis-Besse case, the low pressure safety injection pumps were capable of pumping the fluid that got through the screens. And the high pressure safety injection pump wasn't evaluated, so it is somewhat pump and vendor specific. They did have to modify the high pressure safety

injection pumps for this issue.

CHAIRMAN WALLIS: But one of the commenters said that, if these fibers got through, they would tangle up on things like spaces in fuel.

DR. CHANG: Yes, that's the second comment.

CHAIRMAN WALLIS: They'd be tangling up, and they didn't have to be bigger than the opening in order to start tangling up on these. The spaces themselves are sort of filter or screen. So did you respond to that?

DR. CHANG: Yes. I think that's related to the second comment, as I mentioned -- that the last slivers of fiber may pass through the mesh opening axially and get clogged up later on in those small areas like pump seals or bearings.

So we agree with the comment, and we modified the Reg. Guide to say that people have to consider those conditions if they have that in their plants.

CHAIRMAN WALLIS: Well, you just said consideration should be given to the buildup of debris at downstream locations.

DR. LETELLIER: There is currently a research effort in place --

CHAIRMAN WALLIS: Very vague.

DR. LETELLIER: -- for the next fiscal year to look at screen penetration.

CHAIRMAN WALLIS: Well, there's no criterion for anything. I mean, suppose you say, "Yes, I'm going to have my spacers on some of these fuel elements festooned with fiber," so what? I mean, there's nothing here that says how you decide whether or not it's okay.

DR. CHANG: Well, in this case, we just bring this to the attention of people there. This is a possibility.

CHAIRMAN WALLIS: Well, this whole thing is so vague, you've got to consider all of these things. Are we waiting for some guidance?

MEMBER ROSEN: Yes, the guidance from NEI.

CHAIRMAN WALLIS: Is that what we're waiting for?

MEMBER ROSEN: I think that's the key document.

MR. ARCHITZEL: I'd like to point out that NEI guidance deals with the GSI-191 issue. It doesn't deal at all with the downstream blockage effects. No one -- they're not working on that, so this issue, which is, say, blockage in the fuel channels, is not

part of that effort.

MEMBER ROSEN: Why not? I mean, isn't that -- the Reg. Guide now clearly says "modified to make that comment."

MR. ARCHITZEL: But I'm not saying the Reg. Guide does. I'm just saying it's not part of their current effort.

MEMBER ROSEN: Okay. So now that we've had that comment from the public, and the staff has looked at it and put it -- modified the Reg. Guide, now it seems to me incumbent on NEI to deal with what's now going to be in the Reg. Guide. Am I correct?

MR. LEHNING: This is John Lehning. I guess it's not incumbent on NEI to deal with what's in the Reg. Guide, but it would be incumbent for each licensee to deal with --

MEMBER ROSEN: Right. Well, yes. And the licensees have delegated that to NEI rather than come up with 59 or 69 different solutions, which is logical. So now it seems to me, I mean, you know, we have a coherent system. We have public comment, you respond, you change the Reg. Guide. The utilities now have to deal with what's in the Reg. Guide or come up with alternatives. They don't have to choose to come

up with alternatives.

And they hired NEI to come up with a common method they can use. They set up a task force to work with NEI and to make sure that the guidance comes out the way they think is reasonable and responds to the Reg. Guide appropriately. But, I mean, I'm stunned to think that NEI wouldn't now change the Reg. Guide to deal with this comment, because the Reg. Guide -- change the guidance to deal with this comment, because the Reg. Guide is going to have it in it.

We have an NEI representative here. He could address that. Would you choose to do that?

MR. BUTLER: I don't know what you'd like me to say. I mean --

MEMBER SIEBER: You need to use the microphone.

MR. BUTLER: John Butler at NEI. Our initial effort did not focus on the downstream effects. Part of the difficulty with addressing downstream effects, it's very design-specific, vendor-specific, part-specific. All we could do without an extensive research effort would be to provide some guidance that probably would not go into a lot more detail than the current Reg. Guide point, things that

plants would need to look at to ensure that they've accommodated in some fashion.

But the analyses necessary to demonstrate that their system can accommodate materials that pass through the screens is very pump-specific, design-specific.

MEMBER FORD: Could I ask a question? It's more of a procedural and which I don't understand. This Reg. Guide, this NRC Reg. Guide, gives a lot of qualitative requirements -- assess this, consider that.

Now, do I understand from the conversation that has just gone on that now NEI is going to issue a guidance to their utilities as to how they're going to respond to NRC's request for assessment? So NEI is going to give the quantitative answer?

MEMBER ROSEN: Yes, that's my understanding.

MEMBER FORD: Is that true?

MEMBER ROSEN: Rather than each utility doing it themselves, they've come together in a task force, an NEI task force, which has been charged with the responsibility of coming up with a set of guidance for that -- for each utility to use --

MEMBER FORD: Well, it would close the

circle, then.

MEMBER ROSEN: -- to close the circle.

MEMBER FORD: How will NRC approve, if you like, the NEI's quantification of these requirements?

MEMBER ROSEN: I expect, but I will let them answer for themselves, I expect that they'll read it and write an SER saying that's an acceptable way of meeting the Reg. Guide. Is that correct?

MR. ARCHITZEL: Well, it won't be meeting the Reg. Guide. It'll be an acceptable methodology to address this evaluation that would be addressed in the Generic Letter. But it would be an SER.

MEMBER FORD: And how long will it take to come up with these quantitative guidance to your members?

MR. BUTLER: We're still working toward an end of September schedule.

MEMBER FORD: Gosh. If you read this Los Alamos thing here, I'm not an expert in this area, but you're not looking at a three-month research effort to quantify the interactions between all of these variables to meet their qualitative requirements.

Am I being dumb here? Am I missing something?

MR. BUTLER: No, sir. Let me point out

what Bruce pointed out earlier, that you have a choice in assuming a very conservative assumption or taking a more phenomenological approach to -- that requires a little bit more investigation and detail.

What we're attempting to do with the guidance is provide each utility with options in each phase of the event, as to which method they choose to use. If they can accommodate a very conservative approach in terms of the answer that that gives, that is the simplest and most direct way to get an answer.

In other instances, they will need to provide -- go with a more phenomenological approach, still probably using some conservative assumptions.

MEMBER FORD: Okay.

MR. BUTLER: Because there is not a lot of detailed phenomenological research available that they can use. And there's a large variability in the designs that it would be very difficult to do that on a generic basis.

So the level of detail that they use in their analysis, the level of conservatism they use in their analysis, will be up to each individual plant to meet their needs.

MEMBER FORD: Okay. Thank you.

MR. HSIA: This is Tony Hsia from

Research. I would like to add that the advantage of an issue this -- with such a long history was that industry has done quite a bit already, because we -- that was evident to me when we attended the workshop back in July in Baltimore.

Their plans were to perform analyses to evaluate the debris generation. Their licensee will perform analyses and attempt to figure out a washdown and transport -- washdown from -- you know, with container spray of the debris and transport debris.

And I was impressed to see there was one plant who actually had a very extensive plant walkdown and documented why each room has possible debris. That's the -- later on you will see, when we get into detail, that's -- as it turned out, the NRC and the industry has evolved to really look at this whole thing, and back up a step and say you've got to figure out debris generation, you've got to figure out how to move that debris, whatever you have, from your location down -- washed down to the sump. And this transport in the sump, then eventually the possible blockage of screen and suction strainers.

So that's the direction everybody is going to. And I hope when we get into the detail, you'll be able to see it better.

And I'd also note that Bruce is correct when he said there is no method, he really -- I believe he really meant to say there is no integrated predictive method. In other words, you don't have a code -- let me just put a name out there like a revised RELAP that can include all the debris and predict where they're going to go, and with what kind of force they're going to strike each object. We don't have that tool.

So the best we can do is right now, using codes like RELAP, like MELCOR, at different phases of the accident, and then incorporate that with the test, the knowledge we have gained from experiments on how -- what kind of debris, what size, what kind of debris we'll have, and combine that with the plant-specific configuration. With that all put together, that's the best we can predict today.

So what he meant is there's no integrated simple tool that can give it a solution just by punching in the numbers.

DR. CHANG: Okay. Next slide, please.

The next concern is on the leak before break for the resource. The comment is that Section C.1.3.2 requires application of large breaks in essentially all locations in the reactor coolant

system for regeneration.

This is consistent with 10 CFR 50.46. This is for the calculation of ECCS capability, long-term capability. You have to postulate the most severe postulated LOCAs. But in our case, for the sake of the generation of the worst debris, we used the same approach as 50.46. In other words, they have to consider the most severe postulated LOCAs.

The comment is that this is not consistent with the leak before break position of GDC 4. Our response is there is no change after Reg. Guide. The staff position was documented in a letter to the Westinghouse Owners Group in 2000. The position is that LBB is not applicable to LOCA-generated debris.

However, the staff acknowledges that we have received an NEI request to consider alternatives to a double-ended guillotine break for debris generation. For instance, they postulated maybe we can use the fraction mechanics to predict a certain size of break instead of the double-ended guillotine break.

This is something in between the two extremes. One is the double-ended guillotine break; the other one is the leak before break. So it's sort of a compromise suggestion.

And this is a policy issue which may result in changes to break size used for debris generation. So after we reviewed -- finished reviewing this alternate, what is the status on that now, Ralph?

MR. ARCHITZEL: The last was NEI was going to provide some supplemental material to their earlier application. And once we get that, we plan to go with an ANSI policy paper up to the Commission.

DR. CHANG: Okay.

MEMBER KRESS: Let me ask a technical question, perhaps to Mr. Letellier. How do you pronounce your last name?

DR. LETELLIER: Letellier.

MEMBER KRESS: Oh, you pronounce the R.

DR. LETELLIER: It's been Americanized.

MEMBER KRESS: Is the quantity, size, and transportability of debris in the general locale of the break a strong function of the break size, pipe size?

DR. LETELLIER: The volume of debris is definitely a strong function of the pipe and size. And the correlations are -- have that as a key parameter -- the pipe diameter.

MEMBER KRESS: Okay.

DR. LETELLIER: The amount of debris generated is also a strong function of the damage pressure for the given debris type. As I mentioned, bare insulation to jacketed material to reflected metallic. All of those respond differently.

MEMBER KRESS: Okay. That bears on how I think about this large break LOCA and leak before break issue.

CHAIRMAN WALLIS: I think it's important in big LOCA -- a big LOCA is a really big debris source.

MEMBER KRESS: Yes. But it has a very low probability.

CHAIRMAN WALLIS: That's where the argument is about the leak before break.

DR. CHANG: All right. The next slide, please.

The next comment is on the partially submerged screens, and it's a failure criteria. In the original Reg. Guide sent out for public comment, we have a statement that credit should only be given to the portion of the sump screen that is expected to be submerged at the beginning of recirculation.

Allowance should be provided for circumstances in which the level of submergence

changes substantially following the beginning of recirculation. This is the comment on our statement.

The example cited is it's like using an ice condenser containment, that continually the ice melts and you increase the water level. So if you specify that they have to stick with the water level at the beginning of the switchover, then this is not considered there.

The staff position has been modified in the Reg. Guide to say that for partially submerged sumps credit should only be given to the portion of the sump screen that is expected to be submerged as a function of time. So we added this as a function of time. It's not at the switch of -- switchover time.

Pump failure should be assumed when the head loss across the sump screen is better than half of the submerged screen height, or the NPSH margin. This addresses Dr. Ford's question about there is no failure criteria there. This is the bottom line.

Okay. And originally we have I think -- in the revised version, we have one-half of the pool height. Now we change it to the submerged height. It's because in some designs they have a curb there. A curb effectively is a block of the screen, so you have to count the height without the curb.

Next slide, please.

MEMBER KRESS: That wording is a little strange to me. You're saying that you should assume the pump fails when the head loss across the screen is greater than one-half of the head loss you would get to exceed the net positive suction head origin, or what? I don't -- I'm not --

DR. CHANG: Now which --

MEMBER KRESS: I'm looking at the last sentence of your response. It's just -- I'm trying to read it and see what it actually says.

DR. LETELLIER: Those are two separate criteria. One is the standard NPSH consideration of cavitation at the pump inlet, at the impeller location. You can't violate that margin.

The other criteria is actually a consideration of passing of volumetric flow through the debris bed. The only driving force available is the static head of the water that's sitting in the pool. That's the only way to supply water to the sump well.

MEMBER ROSEN: It would just be a dam that's holding back all --

DR. LETELLIER: That's right.

MEMBER ROSEN: -- the water.

DR. LETELLIER: It's a dam essentially. And on average, your static head is about one-half the pool depth minus the curbing. And so there are actually two separate failure conditions, and I would propose we add the words "whichever is less," the minimum of --

MEMBER KRESS: I just think that sentence needs to --

DR. CHANG: Yes, whichever is less. That's the intention.

CHAIRMAN WALLIS: Actually, they work in combination that -- that you get some drop-in head across the screen, and then you have to worry about NPSH from that lower head. So the two really act together, don't they? They're not independent.

DR. LETELLIER: It's actually the minimum of the two. Whichever is lower will be your threshold.

CHAIRMAN WALLIS: You have to add the two together. Anyway, you'll sort it out.

DR. CHANG: Yes. Next slide, please.

The next comment is on the one-eighth thin bed value. I think we are going to go into the thin bed later on, but the comment is that there seems to be no supporting technical basis to have the number

one-eighth of an inch there in the Guide.

And we made it clear that there is some technical basis in this new --

CHAIRMAN WALLIS: So it's not a magic number. It hasn't really any basis except that it worked for certain physics for certain kinds of material. Nothing magic about an eighth of an inch.

DR. LETELLIER: It is based on test data.

CHAIRMAN WALLIS: No. I mean, it's --

DR. LETELLIER: A bed that's thinner than that will fail.

CHAIRMAN WALLIS: It depends on the screen and the kind of debris and all sorts of things. But anyway, we'll get to that later.

DR. CHANG: Next slide, please.

The next one is on the adequate protection after sump on --

CHAIRMAN WALLIS: This one is an easy one, I think.

DR. CHANG: This is an easy one, I hope.

CHAIRMAN WALLIS: We can pass over this one, unless anyone has a question about it.

DR. CHANG: The next one? Want to skip this one?

CHAIRMAN WALLIS: Well, I had a comment on

this.

DR. CHANG: Oh, you have a comment.

CHAIRMAN WALLIS: In reading the Los Alamos basis -- knowledge basis, it seemed to me that CFD is shown to qualitatively simulate some of these things. But it wasn't really an analytical tool yet.

DR. LETELLIER: Again, the use of CFD codes is to provide engineering information about water velocities and what the transport pads would be. CFD is not sufficient for predicting debris behavior in water. Those models don't exist, and it was not the intent to develop that -- those models.

CHAIRMAN WALLIS: Well, it says analytical -- it's an acceptable analytical approach to predict debris transport. And you're saying it can't do it, so --

DR. LETELLIER: Well, we should clarify that to say when used in combination with test data.

CHAIRMAN WALLIS: Ah, okay. Well, good. Thank you.

Yes. This earthquake one is probably okay, too.

And then we go to slide 17, size of the ZOI. Presumably, Los Alamos has ways to estimate the ZOI that answer this public comment on page --

slide 17.

DR. CHANG: Well, our position is that the ZOI should be consistent with the risk-specific damage pressure. In other words, it should extend until the jet pressure decreases below the experimentally determined damage pressure appropriate for each specific debris source. So this is how it is decided -- the size of the ZOI.

DR. LETELLIER: Specifically, to answer the question directly, to do the zone of influence correlation scale with operating or design pressure, the answer is no. The test data don't exist in a comprehensive fashion. What does exist are zones of influence as a function of damage pressure for the BWR tests that were performed as part of the BWR resolution.

There were limited two-phase blowdown tests conducted as part of this exercise, but not in a comprehensive fashion. What we've done is to account for the difference in the thermal hydraulic conditions and compensate for the difference in energy by reducing the damage pressures. Where for a steam jet, bare, unprotected fiberglass might fail at a damage of 10 psi, we now suggest using a damage pressure of 6 psi.

CHAIRMAN WALLIS: This is a stagnation pressure or what?

DR. LETELLIER: That's right.

CHAIRMAN WALLIS: Okay.

DR. CHANG: Okay. Last slide on the public comment is some samples of other comments. One is on the definitions of NPSH. The one we had in the Reg. Guide before probably isn't too clear, so we quoted the definition from the ANSI document. So it's word by word. It's quoted there.

And the second comment is about the chemical reactions in the pool.

CHAIRMAN WALLIS: I have a comment on that. I mean, all you're considering is the chemical reactions producing precipitate. But on page 120 of the knowledge base document, it speaks about interaction of high pH water with zinc and aluminum surfaces producing hydrogen. And then, later on, on page I31 or 131, it talks about the generation of hydrogen from high pH water.

Now, I've made this point before. When you have bubbles produced on these particles, then you get flotation of the particles. So there are chemical reactions occurring in the pool. There's a continuous bubbling and flotation, rather like the notorious

tanks at Hanford.

And this is going to change the floatability of the debris. And this doesn't seem to be considered at all. I mean, I've made this point three or four times in the past, and no one has ever put it into any Reg. Guide or --

MEMBER ROSEN: Isn't it a conservatism not to consider that? I mean, if the particles --

CHAIRMAN WALLIS: No. Because you have your heavy particle down at the bottom. They throw it away, because it settled.

MEMBER ROSEN: Right.

CHAIRMAN WALLIS: But if it now reacts with gas and makes bubbles, it floats up and gets transported.

MEMBER ROSEN: Right. But it never settles down low enough to go into the pump.

CHAIRMAN WALLIS: It does, because the bubbles fall off essentially. It rises to the surface, the bubbles release, and it falls down again, and goes through a cycle of progressing along and flotation --

MEMBER ROSEN: Well, ultimately, it comes -- it's removed. The bubble is separated from it.

CHAIRMAN WALLIS: Yes, right.

MEMBER ROSEN: But it hits the surface, the bubble separates, and it falls down again. This goes on as long as the chemical reactions go on. You can do it in your kitchen and --

DR. LETELLIER: Two comments. Number one, I'm not sure that the Reg. Guide focuses exclusively on precipitation. I think the words are accurate here that it requires consideration of debris generated by chemical reactions.

CHAIRMAN WALLIS: But it also talks about -- demonstrates that suspended indefinitely or to sink very slowly should be considered to reach the sump screen. It seems to me that stuff which is liable to have bubbles on it and to go through this dance could be considered to be suspended indefinitely.

MEMBER KRESS: I can't believe you're going to produce enough gas in this temperature and condition that it's going to be a significant issue.

CHAIRMAN WALLIS: Show us the --

DR. LETELLIER: That's my second comment is I'm not sure that the scenario that you portrayed is actually realistic.

MEMBER KRESS: Yes. You --

DR. LETELLIER: Keep in mind that the gas generation occurs on exposed metal surfaces. There

are not a lot of exposed --

CHAIRMAN WALLIS: Flakes of aluminum paint.

MEMBER KRESS: Yes. But they don't react -- they're in the water, and this is -- this water temperature is --

CHAIRMAN WALLIS: It says it's got NaOH in it, and all kinds of stuff. It's high pH according to the Los Alamos.

MEMBER KRESS: It's supposed to be high pH to control the iodine problem.

CHAIRMAN WALLIS: That's right.

DR. LETELLIER: The inorganic zinc might be a credible debris source where that should be examined.

CHAIRMAN WALLIS: Well, I don't know. I just assumed that if it's -- if it is a contributor to the hydrogen source term, there must be quite a bit of gas, because there are other contributors. I mean, it's not negligible. It doesn't take much gas to float a particle. Gas has no density at all relative to the water. So, anyway, this should be there somewhere it seems to me.

DR. LETELLIER: I think the focus of hydrogen generation has been on hydrogen deflagration

within containment where your exposed metallic surfaces are impinged by sprays, and the bulk of those metals are not submerged in the pool.

CHAIRMAN WALLIS: Well, they're zinc aluminum paints, right?

DR. LETELLIER: That's true.

CHAIRMAN WALLIS: So they are part of the debris. So I would really appreciate -- and there's aluminum foil in -- crumpled up in this insulation which gets transported, and all that, and it's not something you can just dismiss.

The other thing that there was a comment about that I didn't see on to very well was this business of transient debris. It has been raised by this committee, too, that plastic sheeting, duct tape, and stuff, which happens to be there for maintenance purposes or something, or someone left it there, is simply dismissed as being not something you consider because of risk. Somehow it's considered in the risk analysis. It's not considered as relevant to the screen blockage problem. Why is that?

DR. LETELLIER: No. In fact, it has been considered and excluded based on transportability. Under circulation --

CHAIRMAN WALLIS: That's not the argument

used by the staff in dismissing it, in dismissing the public comment. Maybe there's a physical reason for dismissing it. But they say it's all taken care of by risk, so -- which seems to me very strange.

I have to find it now. Anyway, we can find it. The transient debris public comment.

MR. CARUSO: I'm confused. You said that the sheet material is not transportable?

DR. LETELLIER: Not during recirculation, flows typical of recirculation phase.

MR. CARUSO: On page 2-1 of the knowledge base, it says, "Transportable sheet-like materials, numerous miscellaneous, relatively transportable materials were found that could essentially behave like a solid sheet of material when they're on a strainer screen." Plastic cloth, duct tape, oil cloth, all this -- I don't understand. Are you saying that this is not transportable?

DR. LETELLIER: I hate to mince words. But if you read the recommendation, it says if they are present on the screen, they are of concern.

MR. CARUSO: Why are they listed under "transportable," then? There's another category which is relatively non-transportable.

DR. LETELLIER: There are debris types

that should be considered. And based on your assessment of the fill-up phase, it would be possible to transport that material to the sump, if, for example, the sump represented a very large recessed volume compared to any other location in the facility.

Then, the flows would be preferentially directed towards the screen at a high enough velocity to transport those materials.

MR. CARUSO: This is a pretty simple question, though. You said that they are not transportable, but you've got a document here which says -- which has two categories -- transportable sheet-like materials and relatively non-transportable materials. And non-transportable is hammers, bolts, nuts, stuff that I would expect is non-transportable.

But then you have a category that's called specifically transportable, and it includes all the stuff that Dr. Wallis is concerned about. Is it transportable, or is it not transportable?

DR. LETELLIER: It depends on the velocity regime that you're considering.

CHAIRMAN WALLIS: Well, I guess I'm not concerned about it. It's NEI that's concerned about it, because their public comment says the guidance does not address transient debris sources. Personnel

perform work within containment, so on and so on.

And then, the resolution is dismissal of transient debris sources would be based on risk aspects which have not been otherwise included in the Guide. So they are being dismissed on the basis of risk.

DR. LETELLIER: Well, again, I think we should look at the word --

CHAIRMAN WALLIS: Physically.

DR. LETELLIER: What you read means that if you choose to dismiss these debris, you must have a risk argument to go along with it. I don't think that it implies that those debris have been dismissed with the --

CHAIRMAN WALLIS: Well, does it say that? Does the Guide say that? It just says "disagree." The Guide doesn't seem to address the question at all of transient debris sources.

DR. LETELLIER: Which question number is that, by the way?

CHAIRMAN WALLIS: It's NEI comment number 3.

DR. CHANG: I believe in the record we address those things should be considered as to debris -- let me find it.

CHAIRMAN WALLIS: Well, it's something that we're going to need to look at and resolve. I don't think we can spend the time on it now.

DR. CHANG: I'll try to find it later.

CHAIRMAN WALLIS: I really would appreciate it, some analysis of the hydrogen generation. Even if it's a very small amount, as my colleague says here, then it has to be a very small amount. It's not going to be able to lift up some of these fragments of zinc and aluminum paint.

DR. CHANG: On this chemical reaction issue, the comment is that there is no -- there seems to be no publication out there that NRC published reports of study or cited available references. Our answer is that we acknowledge there are no NRC published references pertinent to this issue that can be cited in the Reg. Guide.

CHAIRMAN WALLIS: So what I'm looking for is a more thorough statement of, what are these chemical reactions in the pool? Other than just debris-generated, what is their effect on the debris? Not just new debris generated by them.

If we have time, Bruce has some slides on the chemical testing, the initial results we have obtained. So we can go them in a little bit.

CHAIRMAN WALLIS: If we have time at the end. I think we're going to run over time anyway, but we -- we probably -- we kind of thought we might anyway. It's an important issue, and we don't have enough time. But we don't have to have a very long discussion at the end probably, so I expect we can adjourn before lunch.

MR. CARUSO: Before you go to the next comment, can I ask a -- this is a naive question about zone of influence. It looks like you only consider double-ended breaks. You don't consider split breaks. Has anyone looked at split breaks at all, zone of influence for split breaks?

DR. LETELLIER: There are correlations available based on the length or the extent of the pressure contour normalized by the orifice diameter. And that would be an appropriate set of data and information to use if you chose to postulate a conical break, like from a fish-mouth orifice.

And, in fact, the NEI is faced with making that choice when they propose a postulated break size based on fracture mechanics. In fact, they have a one-sided jet, and not opposing conical jets that lead to a sphere.

MR. CARUSO: So they just idealize the

fish-mouth, the split break, into a hole of a certain size.

DR. LETELLIER: Yes. Now, keep in mind that the generalization of a --

MR. CARUSO: You can do a round hole of a particular size.

DR. LETELLIER: That's what I mean.

MR. CARUSO: And they don't take into account the geometric effect of a long break as opposed to a round break.

DR. LETELLIER: I believe that's correct. Keep in mind they're trying to establish a compromise between the leak before break, which is essentially a zero damage zone, no appreciable pressure release all the way up to the double-ended guillotine. And so they're looking for a middle ground.

Now, one other point of clarification, the spherical zone is an assumption for convenience, because we don't have predictive models for jet deflections and recollections.

MR. CARUSO: I was just curious.

DR. CHANG: And also, in the workshop in July, I heard that if they consider using the fracture mechanics and considered like it's a hole on the pipe and stuff like double-ended guillotine break, then

they use the hemispherical zone -- use a hemispherical zone.

MR. CARUSO: Okay.

MEMBER KRESS: You're taking the jet cone and finding the pressure that would cause damage to a particular kind of debris out to a certain distance, and that has a volume. And then, my understanding is you're going to make the same volume in a sphere around the pie?

DR. LETELLIER: That's correct.

MEMBER KRESS: That really seems strange to me. I think -- I could go from no -- lots of debris to no debris with that, because you're shrinking the distance of an influence when you do that.

And it seems to me like a more conservative approach would take that distance of the jet influence and draw a sphere at the end of that around the thing, which is a much bigger volume. And that really strikes me as a hokey thing, and it's --

CHAIRMAN WALLIS: But it's the basis of the whole model of generation of debris.

MEMBER KRESS: Yes. And I'm really surprised that we got this one through.

DR. LETELLIER: Keep in mind that the

spherical approximation for a large break LOCA generates a sphere that's over -- between 30 and 40 percent of the containment volume. So even under our current --

(Laughter.)

MEMBER KRESS: That's a lot.

DR. LETELLIER: It is. So if you did what you propose and take the maximum radius --

MEMBER ROSEN: It would be everything.

DR. LETELLIER: -- you would always --

MEMBER KRESS: Yes. Well, I could see that would be an issue.

MR. CARUSO: I mean, I have a very clever garden hose that allows me to dial in different destruction jets. Okay?

(Laughter.)

And I can get very different destructive events, depending on how -- what setting I've got it. Either, you know, a good, solid stream -- it even has this wide flat setting that you can use. And if you're an insect, it matters, you know, whether --

(Laughter.)

-- I have it aimed very carefully, or whether I've got it set on wide destruction.

MR. HSIA: That's why our resident

inspector makes sure they don't have a garden hose like yours in the containment.

(Laughter.)

CHAIRMAN WALLIS: No. But the pipe may have a slit or a hole or -- just like his garden hose.

MR. CARUSO: That's why I asked the question. But we don't consider that. We just consider one round hole, and we vary the size.

CHAIRMAN WALLIS: Ralph?

MR. ARCHITZEL: I just want to make one comment on chemical before you move on. I did want to raise an issue -- it was raised at the workshop -- and that is basically that there's a certain amount of -- if you do get a chance to hear it, you may want to listen to it. But the industry was concerned about not moving forward until there's more knowledge in this area, because they don't know how to address the issue.

So there is a question about timing and resolution of the whole issue associated with chemical precipitation. So you may not need to get into it today, but I'm just pointing out that the industry is concerned and we had indications that until there's more known there's nothing being done to fix this problem.

CHAIRMAN WALLIS: And it's not just precipitation. It's --

MEMBER ROSEN: Well, the implication is if we can find something that we don't know something about, we can delay doing anything forever.

MEMBER FORD: Do we know if industry is moving forward? You say that industry isn't moving forward, or they want --

MR. ARCHITZEL: Well, we have a meeting coming up; we're going to talk to them about it. But the fact is that even our Office of Research isn't taking what's been done any further, so that you can't take what's been done and translate that at the moment into how you do, you know, these complicated analyses, how you factor the precipitation in.

MEMBER ROSEN: Let me just be a little more clear, Ralph. This one ACRS member is not comfortable with the idea that all we need to do is find someone who can ask a question that no one knows an answer to about this, and then we won't have to do anything until that question is answered. I'm simply not -- that is not an acceptable way to work on this problem.

MEMBER FORD: I didn't quite hear your answer to my question, which relates to what Steve is

saying. Is EPRI being proactive on this, and trying to fill in some of the gaps that -- the quantity of gaps in our knowledge?

MR. ARCHITZEL: This is a new issue. I think we've got a meeting scheduled with NEI in September, early September, to try and see --

MEMBER FORD: Well, this is --

MR. ARCHITZEL: -- will they do some research, if we don't, because you need to tie the end of this together. They may be. I think they will be. I'm not sure they're not.

MEMBER FORD: Do they not feel as though it's a high priority item? This has been going on a long time now. They don't see that as a high priority item?

MR. ARCHITZEL: This particular issue -- chemical precipitation -- is a new twist, something that people didn't know about.

MEMBER FORD: Okay.

MR. ARCHITZEL: So they're just being presented with this now as well. It wasn't out there before.

CHAIRMAN WALLIS: Please present them with the whole question of all the effects of chemical reaction, not just precipitation.

MR. HSIA: Tony Hsia from Research. We have been undertaking research on chemical reaction and effective chemical reaction on debris. T.Y. was --

CHAIRMAN WALLIS: Well, let me ask you -- when you have this borated cooler, and you pour in the sodium hydroxide --

DR. LETELLIER: Sodium hydroxide is present in --

CHAIRMAN WALLIS: -- it makes sodium borate, or something like that? What do you make? You must make something like sodium borate? What is that?

DR. LETELLIER: Sodium hydroxide is present in the reactor coolant as a pH buffer, essentially.

CHAIRMAN WALLIS: Well, I'm surprised that you're going to go to a high pH in the pool. It's just because of the iodins, or additional NaOH must be poured in presumably.

MEMBER ROSEN: There is during --

CHAIRMAN WALLIS: To get the high pH -- in other words, you have a low pH from the boron.

DR. LETELLIER: Yes.

MEMBER ROSEN: There's also lithium.

DR. LETELLIER: That's right. And the lithium is --

CHAIRMAN WALLIS: And all of these things interact in some way in the pool and make things which do things, make it slimy or gooey or something. All of this affects the quality of the precipitate of the stuff which is going to get on the screen.

DR. LETELLIER: That's correct. And we are looking at that, and we would be happy to share some of --

MR. HSIA: If you could indulge us to go through the presentation, at the end Bruce had some updated information he would like to share with you.

CHAIRMAN WALLIS: Okay.

MR. HSIA: And I fully agree with Dr. Rosen. I think at this stage we need to move forward with the best knowledge we can, instead of sitting until we solve every single issue, although they important. That's not the right approach from --

CHAIRMAN WALLIS: Well, you can resolve it by being very conservative, I think.

MR. HSIA: Correct.

CHAIRMAN WALLIS: But that might have some real implications for many plants.

MR. HSIA: Correct. Correct.

MEMBER ROSEN: We already have real implications for many plants. We have a question on whether or not we are going to succeed in long-term cooling. That's a significant issue.

CHAIRMAN WALLIS: Which is the last comment. Maybe we can move on to the ACRS comment period.

DR. CHANG: Yes. ACRS, in their letter after the last February meeting, ACRS asked a question that -- because of the susceptibility of sump to debris blockage, other alternative solutions should be looked into to ensure long-term cooling. And the staff was asked to invite the public comments on this issue, and we didn't get any comment from the public on this.

MEMBER ROSEN: The silence was astounding.

CHAIRMAN WALLIS: And actually, it's C.1.2. It's not C.1.1.4. It's C.1.2 -- in my copy of the Guide anyway.

DR. CHANG: C.1.1.4 is about the active sump screen system. So we added that to indicate --

CHAIRMAN WALLIS: This isn't in response to our comments. C.1.2 is in response to --

DR. CHANG: C.1.1.2 --

CHAIRMAN WALLIS: This was supposed to be

a serious comment, and we think this is a problem. It just may not be resolved by analysis of debris transport and all that stuff. It may require that you ensure long-term cooling if the strainers are blocked.

DR. CHANG: Yes. But, again, Bruce has some ideas he wants to share with us --

CHAIRMAN WALLIS: We felt this is a very serious --

DR. CHANG: -- about this issue. He has some slides. Actually, I think you had a handout. You had -- you have two handouts. The other one is on these alternative solutions.

DR. LETELLIER: At this point, there is no substantial information on alternative solutions that we could actually put into the Reg. Guide as beneficial guidance.

DR. CHANG: Just some ideas I guess.

MEMBER ROSEN: Didn't we see one sitting on the floor there at the workshop? I mean, a self-cleaning strainer.

DR. LETELLIER: Yes.

MEMBER ROSEN: I don't understand your point that there are no alternative solutions when one was being offered by a vendor.

CHAIRMAN WALLIS: Wasn't that the point of

that -- I mean, the ACRS comment was that you might have to get water from somewhere else. Wasn't that really our point?

MEMBER ROSEN: Well, yes. But we're flexible enough to realize that maybe even we couldn't perceive an alternative solution that somebody else could. Even us. Even us.

MR. HSIA: But with the leadership provided by ACRS members, we would like to say that our position is we, like Bruce will do later on, we will present some alternative suggestions. But it's really up to the licensee is what -- you know, they have dollars involved. We can be sitting here coming up with very creative fixes, but from an economic point of view they need to cover safety as well as their checkbooks.

DR. CHANG: Regarding the alternative water sources, this is in the Reg. Guide. They can consider alternative water sources as another alternative, if they have the procedure and the training of the operator, and so forth.

CHAIRMAN WALLIS: I think we might move on to the next topic. And I suggest since we're over time -- but I think we're asking questions we would otherwise have asked later in the day, so we may catch

up. T.Y. has part of this next presentation.

DR. CHANG: Right. It's going to be a tag-team approach.

CHAIRMAN WALLIS: If you give your part, and then we have a break before we hear from --

DR. CHANG: It's an alternative. I'll give my part, and then Bruce will chip in. So that's the setup.

CHAIRMAN WALLIS: I don't think we have time to go through the whole thing before the break. But if you can give your part of it --

DR. CHANG: The first --

CHAIRMAN WALLIS: -- then break at a time before Bruce comes in and talks about all the technical matters, then perhaps we can get in the break.

DR. LETELLIER: We intend to address these topics. There are about five separate issues.

CHAIRMAN WALLIS: But it will take quite a long time, won't it?

DR. LETELLIER: It will. We could do the first one, as a suggestion.

DR. CHANG: So maybe we -- let's take the break now and --

CHAIRMAN WALLIS: Take the break now? If

that's what you'd like to do. It's a good break point.

MEMBER ROSEN: It's only five minutes before we're scheduled anyhow, so --

CHAIRMAN WALLIS: Okay. And then we'll try to catch up. But I think we may have to go after 12:00 noon. Just delay lunch. So you've got an incentive to speed up.

Okay. So we'll take a break until 10:25.

(Whereupon, the proceedings in the foregoing matter went off the record at 10:10 a.m. and went back on the record at 10:29 a.m.)

CHAIRMAN WALLIS: We are on the next section.

DR. CHANG: Shall I proceed?

CHAIRMAN WALLIS: Yes, please.

DR. CHANG: The next topic is a summary of our positions in the Reg. Guide, positions and acceptable methods, and also a discussion from Bruce about how those things can be applied in a real plant.

We look at the excellent sequences and it consists of the following: debris sources of generation, then after that you have the debris transport. That includes three types of debris

transport -- the airborne, right after the blowdown, the pipe radiant blowdown, debris is generated, and it can be blown to the containment, and so forth. So this is the airborne debris transport phase.

Then, after the containment spray is turned on, you have washdown debris transport phase. And the sump pool debris transport is on the floor of the containment. You have the flow of all the liquid there, and we have to look at the debris transport in that area, too.

Then, we have a special slide on the sump pool debris transport, and then, lastly, is the collection of all the debris on the screen and what is the head loss because of that.

Next slide, please.

Under the debris sources and generation, consistent with the requirements of 10 CFR 50.46, we have the same words, actually. It says that a number of LOCAs of different sizes and locations should be postulated to provide assurance that the most severe postulated LOCAs are calculated. We've added a few words there for the regeneration calculations.

The original words in 50.46 is for the ECCS cooling and performance calculation. So our thinking is that for consideration of debris

generation you have to be as severe. You have to consider the most severe postulated LOCAs.

And the second bullet is that when we talk about severity, the level of severity should correspond to the postulated break based on potential head loss incurred across the sump pump.

So, actually, this is sort of like -- I think Bruce used the word "break to block." You have to consider the block effect to predict where you have to consider the break.

Then, zone of influence is one of the methods that can be used to estimate the amount of debris generation by a postulated LOCA.

MEMBER KRESS: Now, let me ask you about the first bullet. In Appendix K for ECCS LOCA, they look at the pipe size in postulated, double-ended break here. And the way they vary the pipe size is they look at different pipes that are in the thing, and then -- and break each one of them.

Now, the question that I have about that is, you have a combination, then, of location and pipe size, which determines the severity of the break, and then what is around that particular location.

What's to prevent a big pipe in a given location from having smaller breaks? And is there --

if a pipe is a certain size, is the double-ended guillotine break the most severe for that pipe? So you don't have to worry about smaller breaks in that pipe.

You could do the same thing with -- that they do in Appendix K and just look at different pipes that exist in different locations?

DR. LETELLIER: That's the common practice, to assume that double-ended guillotine break represents the maximum orifice that can be created in a given pipe, and implicitly assume that that is the maximum damage that could be created also.

You don't need to consider small breaks in large pipes unless you need to do a risk analysis where that may dominate the proportion of events.

MEMBER KRESS: Okay.

DR. CHANG: And also, we don't limit ourselves to LOCAs only. If a plant -- the recirculation is needed for a high energy line break, such as main steam or feedwater, then those high energy line breaks should be considered as well. And the most limiting conditions for sump operation should be considered.

And, lastly, all potential debris sources should be considered within a particular ZOI.

CHAIRMAN WALLIS: That excludes any sources anywhere else, such as this floatable plastic sheet.

MEMBER ROSEN: Well, that's for the zone of influence from the break. But the floatable plastic sheet could be someplace else and floated down by washdown, by one of the other mechanisms.

CHAIRMAN WALLIS: Is that considered?

DR. CHANG: Yes. And when you have latent debris and all that --

CHAIRMAN WALLIS: So all that as well. Okay.

DR. CHANG: Yes.

CHAIRMAN WALLIS: I'm sorry. Because I thought it just meant it should be considered only within the ZOI.

MEMBER ROSEN: No, no, no.

CHAIRMAN WALLIS: Oh, okay.

DR. LETELLIER: These are some of the highlights out of the Reg. Guide. We couldn't address every portion.

DR. CHANG: And the next slide, please.

Continuation of debris source and sources and generation. In the Reg. Guide was the position that as a minimum those break locations should be

considered. Perhaps the coolant system -- or main steam and main feedwater, if needed -- with the largest amount of potential debris within the postulated ZOI.

And the next one is large breaks with two or more different types of debris within the expected ZOI. Breaks in the areas with the most direct path to the sump. I think that's obvious.

And then, the last two -- I think they are interrelated. It's about the thin bed effect. So the break with the largest potential particulate to the insulation ratio by weight should be considered.

DR. LETELLIER: Now, the next slide tries to address or introduce you to the acceptable methods. Now, we talked about a number of these back in the February subcommittee meeting where I went through a rather exhaustive survey of each phase of the accident sequence. But I felt that it was necessary to -- or useful to reemphasize some points that T.Y. has made.

In order to assess so many different suggested break locations, some sort of spatial model or drawings, information about your plant, is essential. And, in fact, at the workshop we saw where the plants are making progress at reconstructing that information where it did not exist before. Some

plants already have three-dimensional CAD inventories of their insulation and piping systems.

The methodology that they choose for assessing the various locations is entirely up to them. There is always the conservative approach of 100 percent damage, if that's a tenable solution. Otherwise, some sort of mechanized, systematic survey may be necessary.

Essentially, we're interested in postulated breaks in all systems that lead to a recirculation requirement. That is the scope of GSI-191, long-term cooling. And so main steam line breaks, for example, or steam tube ruptures can lead to a requirement for recirculation in some plants.

The third bullet -- having a definition of break severity that's defined in terms of a potential head loss, that implies a break to blockage transport analysis, even if it's done only crudely with transport fractions -- 50 percent, 70 percent.

You have to be able to assess the impact of a postulated break on the eventual head loss. That's the reason, for example, that pipe size alone, as defined for the purpose of cooling capacity, is not the single criteria.

CHAIRMAN WALLIS: So that means that if

you have, say, a big pipe, you consider different places where it might break, and you consider if it broke here, there's more debris in that area, although it's got a zone of influence that's humongous no matter where it is. But it would be worse to have it here from the point of view of dislodging stuff.

DR. LETELLIER: For small breaks, that's more likely to happen, because the zone of influence is smaller. And in some plants, we've noticed that there is more small piping in the vicinity of problem debris, for example. That's the rationale that we use to add the words for maximum number of debris types, for example.

As far as acceptable methods go, we've mentioned the 100 percent criterion, and that's always an option that we won't dwell on. However, there is a precedent in both NUREG-6224, which was the cornerstone document for the BWR resolution.

It sets a precedent for a point-by-point break analysis, where we proceed systematically through all of the piping systems and examine many hundreds of potential breaks. That is a method that's familiar to the staff and would be deemed acceptable.

Now, that's not to say that this is a requirement for every plant. The spatial details may

be simplified, for example, by considering the plant-specific insulation applications -- predominantly, RMI plants, reflective metallic, may not have to do as exhaustive a search for break locations. They may focus primarily on the areas that include the fibrous material, some residual material. And there's a wide variety of plant configurations.

The next slide, 23, points you to some specific references to address the panel's interest in peer review. I think you've got the impression that we have shared our research findings with industry in a participatory fashion for many, many years, both at the local and international levels.

It's very difficult to point to examples where a formal peer panel was convened in a formal process. But there have been a number of important opportunities for critique and criticism, and they're listed here.

For debris source references, there was an early survey of insulation types used done in 1981. More recently, in response to Generic Letter 97-04, the NEI conducted a plant-wide survey that compiled a list of industry responses to specific questions asked by the staff.

And the knowledge base reference will come

up repeatedly as a blanket document. It's the most recent compilation of research findings that has been subject to international critique. And, again, we could look at the comment resolution history and make a judgment whether that was adequate in the committee's opinion. But, in fact, it was open to everyone's input.

CHAIRMAN WALLIS: It seems to me that the regulatory process cannot be independent of the knowledge base. If the knowledge base is very precise, you have a certain kind of regulatory process. If the knowledge base is extraordinarily vague, then you're going to have a different appropriate regulatory process.

I think one of the things that concerned me was that the -- there seemed to be -- these didn't seem to be the right -- didn't seem to have the right connection. The Guide is asking for all kinds of calculations. The knowledge base doesn't let you do it.

If the Guide was more acknowledging that you couldn't do things, and said that you should assume other things, then they might fit together better. I think that's a concern I have.

DR. CHANG: The attempt here is trying to

establish the link as far as we can in this presentation.

MEMBER FORD: But following up on Graham's comment -- this is really a tag-team act here -- this is an excellent source for the utility to go away and find out, well, what sort of debris sources should they be worried about?

But practically, surely the debris source that they should be worried about for their specific plant will depend on details of the break, type of -- whether it's spherical or what sort of break it is, if it depends on the various transport mechanisms for the specific debris.

So you just can't take this by itself. Is that a true statement? And so this knowledge is not enough --

DR. LETELLIER: Debris source is --

MEMBER FORD: -- to satisfy some of the requirements in your Regulatory Guide.

DR. LETELLIER: That is correct. And that's why I emphasized the philosophy of a break to blockage analysis. You have to integrate all steps, all phases of the accident sequence before you can decide whether you've found the most conservative or the bounding event.

MEMBER FORD: But I don't see the -- in your report here, knowledge base report, you talk about quite specific -- this, then this, and then this. They're not tied together. Is that true? Is that a true --

DR. LETELLIER: That's a fair observation. Now, the document that will come up here shortly, NUREG-6224 represented an integrated analysis of a BWR vulnerability assessment. That's the best template that we have for the end-to-end consideration of effects.

There has been an ongoing project in the NRC to conduct a volunteer plan assessment that would have provided a very similar example of how to apply the integrated assessment. Various priorities have pushed that aside for the moment. But I'd have to say that even the volunteer plan assessment relied very heavily on 6224, and that is available.

MEMBER FORD: Now, why aren't the utilities doing all of this work?

DR. LETELLIER: Ultimately, they will. Ultimately, each utility will have to conduct a similar assessment.

MEMBER FORD: I'm talking about the utilities as an industry, as a conglomerate. This is

a generic problem. So why aren't the utilities --

DR. CHANG: Well, they are trying to come up with utility guidelines through the NEI. So there's a general document that -- I think they are talking about at the end of September, will they have that document ready for us to review.

MEMBER FORD: And the information is there, so they can come up with an integral -- integrated approach to this?

DR. CHANG: Hopefully.

DR. LETELLIER: Their guidance will be based heavily on the knowledge base and what's available in the literature. I guess maybe a personal concern is that the knowledge base is not comprehensive. It does not address all of the materials of potential concern.

MR. ARCHITZEL: We'd like to point out that even though we're going to get that schedule now in September, we have had ground rule documents over the last four or five months on some of the areas. So they have been doing something. They've given us some high-level type information as to how they plan to address this. So it's not like they're just starting this month. They --

MEMBER FORD: Okay.

MR. ARCHITZEL: They have been looking into it.

DR. LETELLIER: So let's go on to the associated consideration. The zone of influence -- and we've already talked a bit about this. Maybe I should simply ask for questions to clarify our assumptions of the spherical zone of influence.

Keep in mind that it is dependent -- the correlations are dependent on the break size, and the damage pressure of the debris type you're interested in.

CHAIRMAN WALLIS: Well, let's see. This is a model. Have there been tests that show that using a spherical zone of influence with the sorts of piping you might get and the sorts of pressures you might get and the scales you might get actually work reasonably well?

DR. LETELLIER: There have been some tests with double-ended guillotine, with no offset, with complete separation but no offset, that show that opposing cones tend to deflect in a roughly spherical manner.

And the argument perhaps more appropriate for the BWRs is there is so much piping congestion that the random deflections will lead to a zone

roughly like a sphere. It's an assumption made for convenience. It does not account for the loss of energy during redirection of the jet. It essentially maps the pressure contour from a free jet expansion into an equivalent volume sphere.

CHAIRMAN WALLIS: Why isn't the pressure everywhere -- stagnation pressure, bring it to rest?

DR. LETELLIER: These are the stagnation pressures that would occur against a blockage.

CHAIRMAN WALLIS: So it must -- there must be some dissipation or something of energy out there. If you typically take a flow coming out of a pipe isentropically, and then bring it back to rest again, it goes back to the pressure it started at. So something must happen to disperse it.

DR. LETELLIER: I'm not sure that I understand the question. You're talking about free field expansion and --

CHAIRMAN WALLIS: Well, if I take my colleague's garden hose with a pressure of 40 psi, or 50 psi, let's say, g, and I direct it at a wall, I get 50 psig on the wall, unless there's some kind of losses in the flow.

DR. LETELLIER: These are freely expanding gases that are expanding into a lower pressure.

CHAIRMAN WALLIS: Yes. But then, when you compress them again, they go back to where they started from, unless there's some dissipative mechanisms.

DR. LETELLIER: Well, the dissipative mechanism is partly geometric as you expand.

CHAIRMAN WALLIS: Well, I don't think that works out, though.

MEMBER KRESS: I don't think you expand isometrically.

CHAIRMAN WALLIS: I said it's isentropic.

MEMBER KRESS: Yes, it's isentropic.

CHAIRMAN WALLIS: There must be some losses there.

DR. LETELLIER: Yes. You can't expand isentropically.

CHAIRMAN WALLIS: Why not?

MEMBER KRESS: Somewhere in between the two.

DR. LETELLIER: The damage pressures were actually based on test data where they had witness objects positioned at various points in the jet, so that the damage pressures could be correlated to some of the ANSI and ANS jet models at the -- under acceptable methods at the bottom of this slide, it

lists some of the tools that are available.

For example, the industry is interested in redirecting the jet to -- I guess to alleviate the limitation that we're ignoring concrete barriers essentially. There is no jet deflection, no truncation due to walls.

But what they would like to attempt is to remap the equivalent pressure volume into the compartments where the break occurs. And to do that, they will need access to tools like the ANSI/ANS model.

MEMBER RANSOM: What kind of tool did you say?

DR. LETELLIER: There are models available for free jet expansion.

MEMBER RANSOM: Free supersonic?

DR. LETELLIER: Right. To look at the shockwave generation. Two of those are mentioned by -- reference ANS in the EPRI jet model.

MEMBER RANSOM: I guess one of my comments would be the -- you know, a free jet even is very non-uniform in terms of the -- it doesn't have spherical profiles in it. And it actually has shocks in it caused by the ambient pressure and compression on the boundary.

And I'm wondering if actually a better model for the damage is the dynamic pressure, one-half fluid density squared, which varies somewhat from the stagnation pressure. But generally, it's, you know, what dictates drag and --

CHAIRMAN WALLIS: Does he know what he's talking about?

MEMBER RANSOM: It's something close to the stagnation pressure --

CHAIRMAN WALLIS: I think he's talking about the -- that the pressure you measure is the -- bringing this stuff to rest on a wall or something.

MEMBER RANSOM: Well, it's just the stagnation pressure.

CHAIRMAN WALLIS: Well, it's --

MEMBER RANSOM: Minus whatever you get in a shockwave basis.

DR. LETELLIER: I believe I'd have to do some more homework to give a specific answer to your question about the form of the model. I did want to point out that the precedent for a spherical destruction model was introduced very early, before 1985, as part of the USI-A43 resolution, where they postulated zones from complete damage to partial damage to zero damage.

And as data have been added to the knowledge base, this has been refined into a correlation. Again, the correlations are based on pressurized air surrogates for steam. And there were limited tests done for GSI-191 looking at two-phase jet expansion.

Unfortunately, the test data that was obtained was not extensive in scope. It was performed for a lower operating pressure and a smaller volume. And so scaling arguments were invoked to compensate for those differences, in order to adjust the assumed damage pressure of each insulation type.

CHAIRMAN WALLIS: I'm trying to think about the difference. If you have an explosion, and you get something like an acoustic wave which goes out, and that attenuates with area --

DR. LETELLIER: Right.

CHAIRMAN WALLIS: -- because it's not the same stuff. I mean, it's a wave going through, and the gas which is out here isn't the same as the gas which was in here. But when you have a flow of stuff coming out of a pipe, it goes out like a hose and it hits something, and unless that flow of stuff loses some mechanical energy on its way, it's going to have the same energy it started with.

DR. LETELLIER: Certainly, there are mixing processes on the boundary of the wave that --

CHAIRMAN WALLIS: So I think the spherical thing may well have originated from an analysis of explosion.

DR. LETELLIER: The assumption of a spherical zone is a practicality, just based on the uncertainties of deflection in a congested piping environment.

CHAIRMAN WALLIS: But if I'm a policeman with a hose trying to control a crowd, I don't want a spherical zone of influence. So, you know, you -- it's obviously a big assumption which -- and your reply about the empirical evidence seemed to be that for a certain kind of a break you could make -- map pressures in some way. And it seemed that they were roughly in a spherical pattern around the hole.

But did it show that if you used these pressures for damage calculations, you got the right answer, too? The synthesis of the spherical model with the damage, showing that you've really got the right pressure and damage with your model, other than just the pressure itself.

DR. LETELLIER: I think there are many acknowledged deficiencies to the assumption. But keep

in mind that the purpose is to estimate or to conservatively estimate the maximum --

CHAIRMAN WALLIS: Well, look at Barseback. Barseback had a relief valve or something that popped, sent out jets of steam. Was the damage in the direction in which the jet went, or was it in the sphere? There must be some evidence there. You saw a description in your book here about all of these events. Did anyone go in and say, "These events show that there really was a spherical behavior," or not?

DR. LETELLIER: That's a very good question.

CHAIRMAN WALLIS: Well, I mean, that's the kind of question I have about all of this. There's the description of things that happened, and then there's somebody's thought model of what might have been a good way to represent it. And what's the connection?

MR. HSIA: Chairman Wallis, Tony Hsia from Research. I believe that the -- one of the reasons we proposed the spherical model as an alternative is to take into consideration the conservatism, because if you say the directional -- jet has a certain direction, and hits an object, then the argument would be, well, how do you know it's going to

disorientation? How do you know it's not going to start with the jet going to the 90 degrees from this? So there's no end as far as which direction you should point the jet at.

So in order to cover that, we felt the spherical model -- as long as you have a break at that location --

CHAIRMAN WALLIS: But we don't think it is conservative, because the sphere attenuates.

MR. HSIA: So does the directional jet.

CHAIRMAN WALLIS: Yes, but not so much in the direction in which it's going.

MR. HSIA: Well --

DR. LETELLIER: We have not accounted for the attenuation of an actual spherical release. What we've done is assumed the free jet expansion that does have a characteristic pressure gradient, with no deflection, and we've remapped the equivalent energy into a sphere.

CHAIRMAN WALLIS: But if my obnoxious grandson wants to spray his charming cousin with a water jet, he aims the jet at the person. He doesn't put out a spherical jet, which would be useless. It would just be a gentle little mist and sort of around -- it's different.

MEMBER RANSOM: Let me try something and see if I understand what they're doing. It may explain your problem with this, too, Professor Wallis.

I think that, you know, the highest mark numbers are found along the centerline of the jet in a free jet. And those are the areas of highest dynamic pressure. And, of course, as you pointed out, the stagnation pressure is going to be constant along that. So it's all equal to whatever it was in the pipe.

Now, they have to assume a damage model, and worse damage is going to occur along the centerline of that jet. So I think what they've done is they simply said, "Okay. We're just going to take a hemisphere or a sphere and assume everything in that area is going to be damaged all along the centerline of the jet."

CHAIRMAN WALLIS: It doesn't, because they attenuate the pressure. They don't --

MEMBER RANSOM: No, no, they don't.

DR. LETELLIER: No, we don't attenuate the pressure.

CHAIRMAN WALLIS: You don't keep the pressure all the way out to the --

MEMBER RANSOM: Well, they do not preserve

continuity and assume the flows through the spherical areas, I don't --

CHAIRMAN WALLIS: Well, they do, because the zone of influence is bigger for certain things than others. So there's a bigger pressure closer to the hole than there is further away.

MEMBER RANSOM: There's a bigger pressure where?

CHAIRMAN WALLIS: Closer to the break. They have a sphere for radiative, reflective, metallic insulation. We need the picture. And then, they have a sphere for calcium silicate and a sphere for fiberglass. This is because the pressures are getting less as they go out from --

MEMBER RANSOM: Well, that would be true of the static pressure, but not the stagnation pressure.

CHAIRMAN WALLIS: Well, but that's I think the question we have with us.

MEMBER RANSOM: Then they've got something screwed up.

CHAIRMAN WALLIS: This is an acceptable method.

DR. LETELLIER: The final point I'd like to make -- your analogy about a directed jet being

more effective. That depends very much on the uniformity of your target. If you're concerned about a point target at some distance away, the directed jet is more effective.

But the compromise, the practical compromise was made that debris targets in congested piping system, they exist all around you. And that it's an acceptable approximation to map a sphere to --

CHAIRMAN WALLIS: Well, is it? Because I have the 15,000 cubic feet of fiber measured in the air handling units. And normally they would be quite a long way away from this hole, I think.

DR. LETELLIER: And, again --

CHAIRMAN WALLIS: But if I had a directional jet aimed at an air handling unit, it would presumably dislodge 1,000 cubic feet of fiber.

DR. LETELLIER: I don't think that the data support that. Even stainless steel jacketed fiberglass insulation can be quite robust.

CHAIRMAN WALLIS: Not against the stagnation pressure of one of these jets -- 2,000 psi?

DR. LETELLIER: Yes.

CHAIRMAN WALLIS: Yes?

DR. LETELLIER: The damage pressure changes from unprotected fiberglass damage pressure of

10 psi. You can achieve 140 psi damage pressure.

CHAIRMAN WALLIS: Well, I'm talking about 2,000, if we conserve the stagnation pressure, in the extreme case of a directional jet.

MEMBER ROSEN: That's a strong jet. Get hit by a 2,000 psi something, there isn't much insulation that could stand up to that.

MEMBER SIEBER: Well, with the exception of main steam and feedwater piping, most of the high energy lines are in cubicles where there is a physical boundary surrounding wherever the leak may be. And in that cubicle will be things like reactor coolant pumps, steam generators, other valves, other pieces of piping, small bore lines.

And I would think that with all of these obstacles in that small space that the assumption that a single directed jet would -- just wouldn't fit physically.

CHAIRMAN WALLIS: Old Faithful is a break in the pipe. And it doesn't have a spherical pattern.

MEMBER SIEBER: It doesn't have a lid on it either.

CHAIRMAN WALLIS: No. But you know it --

DR. LETELLIER: And it doesn't extend indefinitely. There are dissipation processes that --

CHAIRMAN WALLIS: No. But it's a focused jet, and the attenuation of that jet is not anything like as rapid as it is if you work it out from Surrey.

DR. LETELLIER: But keep in mind, again, the damage pressures were based on free jet expansion of experimental configurations where you had pressurized air with a perforated nozzle, perforated plate. And so those experiments do incorporate realistic dissipation mechanisms, and we are not taking credit for --

CHAIRMAN WALLIS: Was this pressurized air?

DR. LETELLIER: It was indeed.

CHAIRMAN WALLIS: Because if it's water, then it should keep going the direction it started in.

DR. LETELLIER: That's correct, and that's the reason I pointed out the distinction between the two-phase blowdown test. The database is quite limited, but we do understand what some of the discrepancies are. And we've tried to compensate accordingly.

Next topic?

DR. CHANG: The next topic is about the debris transport. In the Reg. Guide, we stated that debris transport analyses should consider each type of

insulation and debris size. And the three types of debris transport should be considered. They are airborne, washdown, and sump pool debris transport.

And one conservative approach that is acceptable to the staff is that instead of doing a detailed analysis of those transports, one can simply assume that all debris will be transported and collected at the sump screen.

However, if all screens -- if all drains leading to the sump could become blocked, or eventually can be held up -- and that could happen in conjunction with the debris on a screen -- then the consequences could be worse than 100 percent debris transport to the screen. And this scenario has to be assessed as well.

So assuming all the debris are transported to the screen may not be always the worst case.

CHAIRMAN WALLIS: This is where the plastic sheet may come in, and blocking a drain it -- if it were close to the drain already, it might not have to move very far.

MEMBER ROSEN: This is where you don't get any water in the sump at all.

DR. CHANG: Right.

MEMBER ROSEN: Right, right. You just get

air into the pipe, right? Do you --

DR. CHANG: This is completely blocked. The water level is very low.

MEMBER ROSEN: And the valves still get a signal to open, and the pumps get a signal to start, and all you get is air.

DR. LETELLIER: That's correct.

MEMBER ROSEN: Yes. Yes. That's what's going to -- should be analyzed here, right? That air ingestion?

CHAIRMAN WALLIS: Well, it doesn't cool the reactor.

MEMBER ROSEN: No. It does worse than not cool the reactor. It completely binds up the whole safety system.

CHAIRMAN WALLIS: Well, I think you don't want to inject air into a hot reactor anyway.

DR. LETELLIER: If you have no water in the sump, but then you violated your NPSH margin, you have no --

MEMBER ROSEN: Right. But I'm saying, couldn't it be worse than that? I mean, now you've got -- the analysis I assume you're asking for here is if you get no water in the sump, what really happens? Including air ingestion into the suction of the ECCS

pumps.

CHAIRMAN WALLIS: Maybe the worst might be the --

DR. CHANG: There's always some water in the sump. But the sump level may be not as we expected, because --

MEMBER ROSEN: There's always water in the sump? How is that?

DR. CHANG: Because of the break -- flowdown of break flow, and now also containment spray.

MEMBER ROSEN: Are you assuming here it's all 100 percent blocked?

DR. CHANG: No. The block is the drain -- drain blockage.

MEMBER ROSEN: Okay. So you're going to get water some other way.

DR. CHANG: Right.

MEMBER ROSEN: Not through the drains, just washed --

MR. ARCHITZEL: I don't think we asked for that to be analyzed, I'm pretty sure. Maybe you don't understand the bullet correctly. You analyze it to prevent it from happening. You don't -- we don't have a design basis accident with the sump inoperable, so

you need enough NPSH, and you fix it if you don't have it.

But you don't sit there and analyze the condition where you don't have NPSH, where you have no water in the sump. That's not what we asked utilities to do.

CHAIRMAN WALLIS: You just decree it can't happen.

MR. ARCHITZEL: We asked them to make sure to analyze it, so it can't happen.

DR. LETELLIER: We're using NPSH margin as the threshold of concern. If you've lost margin, then we effectively assume that you have no capacity for long-term cooling.

MEMBER ROSEN: What does this statement in the last bullet on the slide that the consequence could be worse than 100 percent transport mean?

DR. LETELLIER: If, for example, that you had a screen design that was capable of accommodating 100 percent of the debris -- of the insulation inventory, with acceptable head loss across that bed, it would be far worse if you had an alternative condition that blocked all of the drainage paths and prevented water from reaching --

CHAIRMAN WALLIS: And you have a dry sump.

DR. LETELLIER: That's correct.

CHAIRMAN WALLIS: Right. And that's something that is of concern.

MEMBER ROSEN: That's right. And so you have the dry sump. Now I ask, what happens then? I mean, is that a legal question?

DR. LETELLIER: If you have no water, you have no margin. And so that's, in effect, a regulatory failure. We're not concerned about the consequences or the progression of that event.

MEMBER ROSEN: Okay. So it gets worse, but you don't -- you already lost the game 56 to nothing.

DR. LETELLIER: That's right.

MEMBER ROSEN: So why do you care if you lose it 65 to nothing?

DR. LETELLIER: That may be a legitimate concern for recovery of mitigation options, but not for the purpose of regulatory guidance.

MEMBER ROSEN: Okay.

MR. ARCHITZEL: That would be in severe accident spaces.

MEMBER ROSEN: It's a worse severe accident space consideration perhaps, but it's not a -- we're not talking about that yet.

MEMBER FORD: But is there any mechanism to toss that concern onto some other group? I mean, you're drawing a firewall down this particular situation. And you're saying, "Okay, I'm not considering that part." Well, who does consider that part? It's a communications issue, isn't it? I mean, who is --

CHAIRMAN WALLIS: NRR.

MEMBER FORD: What?

CHAIRMAN WALLIS: NRR.

MEMBER FORD: Well, yes. But I'm hearing, "No, we're not going to consider that."

MR. ARCHITZEL: I think design basis space in the Reg. Guide. But as far as severe accident goes, we have another branch that looks at -- they include failure of sump for different reasons.

MEMBER FORD: That would already be covered. That's already covered.

MR. ARCHITZEL: That's assessed outside of design basis accident. We're using this Reg. Guide for DBA analysis. We're not using the Reg. Guide for severe accident analyses. That's another group that looks at -- sump failure is one of the things that happens. How do you mitigate? It's probabilistic, it's a Level 3, it's not our group at all that looks

at that.

MEMBER SIEBER: Well, this Reg. Guide is designed to provide an acceptable methodology to show that you comply with the GDCs, which specify that you ought to have recirculation capability. And so the other side of the question is, you know, if you don't comply, of course, you don't comply. And there's a problem; you ought to be shutting down.

But if you don't comply in the course of an accident, you're into -- beyond the design basis space and emergency planning and all kinds of things like that -- severe accident.

MEMBER ROSEN: So I guess the answer to your question, Peter, is that somebody else will look at the implications of this in severe accident space and consider one of these SAMAs they call them -- severe accident mitigation alternatives. And that the SAMGs, the severe accident mitigation guidelines, will somehow take note of this at some point and be revised. Is that what I'm hearing?

MR. ARCHITZEL: I'm not sure there's anything active in that area. I'm just saying it currently is an area that's examined by severe accident management guidelines and evaluated. Sump failure is one of those.

For example, when Davis-Besse came up and it was evaluated, you know, the -- what about it -- if the sump had blocked because of this hole in the head, you know? And then, it was evaluated by the PRA staff about, you know, you flood up around a vessel. And, yes, you don't have any recirculation, but you can have cooling that way. It is a potential to get onto --

CHAIRMAN WALLIS: My question was purely prompted by essentially a question of procedure.

MR. ARCHITZEL: Yes. I don't think there's anything active.

CHAIRMAN WALLIS: But someone is looking at it.

MR. ARCHITZEL: I don't think there's an active look at this.

MEMBER ROSEN: Well, you heard it here. Right, Ralph? Tony? You heard it here that someone thought, well, if it's as bad as that, what can -- innocent question, what happens then? And you need to think -- and your answer is, "Well, it's considered in severe accident space." And we tell you, "Okay. Pass that along to the severe accident people."

MR. ARCHITZEL: Right.

MEMBER ROSEN: Let them do so.

MR. ARCHITZEL: Right.

MEMBER SIEBER: Yes. But this is not a new issue. That was done 20 years ago -- severe accident --

MEMBER ROSEN: That may be the answer that the severe accident people tell us. It's not any worse than something we've already considered, so it's fine. That's okay. I don't want to make a big deal of it. I just want to understand the process.

DR. CHANG: To clarify one thing, I think, Dr. Rosen, when you talk about sump, I think there's a confusion of terminology. We use the sump pool as the floor of the containment. I was referring to the sump pool there as there will always be -- there is always going to be some water, whereas the sump you are referring to is the pit. Okay. So the dry pit is a possibility.

MEMBER ROSEN: Yes, I'm worried about a dry pit where the suction -- the end of the suction piping is.

CHAIRMAN WALLIS: Okay. Can we move on now, then?

DR. CHANG: Okay. Bruce, it's your slide.

DR. LETELLIER: To discuss briefly what methods are available to assess the transport during

blowdown and washdown, the only method -- systematic method for doing this at the moment is to combine some information about updraft velocities and water drainage pathways with information about transport characteristics of debris types.

And the method that's been applied for the BWRs is this logic chart. It's essentially an event sequence that maps the disposition of various debris fractures -- the large pieces, the small pieces of each insulation type throughout containment.

We've actually used the code MELCOR to get some impression of the updraft velocity through the various compartments, what portions of the flow expand throughout containment, in order to make some informed judgments about what fraction of debris are transported.

CHAIRMAN WALLIS: This is slide 27, then? Is that -- you need to move this one.

DR. LETELLIER: My apologies. Thank you.

Ultimately, these judgments have to be made from the point of view of conservatism. If you are attempting to rationalize a washdown fraction of five percent, then you need to have supporting evidence to do that.

We've done a very detailed examination of

our volunteer plant, and we find it difficult to argue for less than 60 percent transport back to the pool. Keep in mind that there is some initial impingement on the floor. Some portion of the debris will impact the floor and be available during pool fill-up.

The bulk of the fine debris will be lofted throughout containment, but it will be small enough to be entrained in condensation flows and spray -- through spray washdown.

So this logic diagram was vetted -- first vetted in 6224 as part of the BWR resolution. I should state that as a cornerstone document the 6224 was preceded by a PIRT review, so that they prioritized the appropriate phenomena.

The PIRT was reconvened at the end of that study. I'm sorry. I misquoted the reference. They were reconvened to examine the drywell debris transport study, which implemented this method. And so it has had a peer review in that context.

Again, a similar statement -- there are no integrated numerical models that are appropriate for transport of specific debris types. We have to combine flow velocity potential with transport characteristics.

CHAIRMAN WALLIS: You've said that it was

difficult to argue for less than 60 percent of the debris being transported to the sump? Well, if that's a bottom possibility, maybe 80 percent is more realistic, and you might as well assume 100 percent to be conservative.

DR. LETELLIER: The purpose of our examination was largely to offer some recommendation whether that's cost effective to do, whether you choose to construct a phenomenology model to gain that advantage or not.

The next slide shows the references that are available. I've already mentioned volumes 1, 2, and 3 of the drywell debris transport study and the application of this method to the BWR resolution.

CHAIRMAN WALLIS: So now each plant is going to develop, based on this knowledge base, its own method? I think there's going to be huge diversity unless they fit up with an NEI guidance or something.

DR. LETELLIER: I expect that in large portion they will adopt the NEI guidance.

DR. CHANG: The next slide is about the sump pool debris transport. We stated that this transport should include debris transport during -- for fill-up phase and the recirculation phase, and

also, the turbulence in the pool caused by flow of water, water entering the pool from the break flow, and containment spray vent drainage. Those are the water sources.

And thirdly, the buoyancy of the debris should be considered also.

CHAIRMAN WALLIS: Which includes mixtures of debris.

DR. CHANG: Right.

CHAIRMAN WALLIS: Including gas maybe.

DR. CHANG: For instance, if the debris is not broken down, if there is air trapped, it may be floating. But as the time goes on, if it disintegrates, then it would make -- eventually settle down to the bottom.

CHAIRMAN WALLIS: Yes, maybe.

DR. CHANG: Yes. So those things should all be considered.

Also, the debris that should be considered in the transport analyses are -- that float along the pool surface, that may remain suspended during the pool turbulence, and also those readily accessible to the pool force. So all sorts of debris should be considered in the transport analysis.

And I think we got this last bullet right.

We said CFD assimilation in combination with experimental debris transport data is an acceptable approach. So we are having to modify the Reg. Guide in this -- in those words.

And we also mentioned that alternative methods would be acceptable. I think this is a general statement true for the whole Reg. Guide, if they can be supported by adequate validation of analytical techniques using experimental data to ensure that the debris transport estimates are conservative with respect to the quantities and types of debris transported to the sump screen. Okay.

DR. LETELLIER: And the practical applications of this guidance are discussed next on slide number 30. When I made the statement before about 60 percent transport, that was specifically with regard to blowdown and washdown. So we're talking about 60 percent of the generated volume being introduced to the pool or at the floor level.

The additional fraction that's lost from pool transport is largely dependent on when and where it arrives in the pool. Debris that's impacted on the floor is subject to fill-up flow velocities, which can be very high, and they are very directional depending on the plant geometry.

That's probably the best opportunity for sequestering debris in quiet sump areas. For example, many containments have opposing steam generator compartments. If a break occurs on one side, the opposite compartment is very quiet and does not participate in directed flows.

Some portion of the debris will find its way into those areas. Elevator shafts and, in particular, reactor cavities also represent dead zones with significant potential for holding up debris. Before credit can be taken for those areas, some consideration has to be given to the drainage flow paths.

In our volunteer plant, we identified between 8 and 12 locations where you would be dropping between 500 and 1,000 gallons per minute in a fairly localized area. That's a significant source of energy of turbulence in the pool. And so there are phases with regard to the velocity pattern.

The picture that's shown is intended to represent the steady state flow velocities where the cylinder in the steam generator compartment is the source of the break, and the sump is near the bottom of the annulus. So this is sort of a steady state configuration that would persist for long term.

MEMBER ROSEN: The red is high velocity or low velocity?

DR. LETELLIER: The red is any velocity exceeding .2 feet per second. That's sort of a rule of thumb for transportability of various debris types.

MEMBER ROSEN: So anything in the red zone will transport. Anything in the blue/green zones will probably not transport.

DR. LETELLIER: That's correct. There is a potential for transport anywhere within the red zone. These patterns are very plant-specific. For example, our volunteer plant has elevated steam generator compartments, so there's essentially concrete inside of these cavities that cannot participate in the sump pool. They're excluded.

So, essentially, the annulus is the only volume where debris will reside. And that's a condition that's very vulnerable to additional debris degradation from --

CHAIRMAN WALLIS: So if it's in this region of greater than .2 feet a second, it's up in suspension, and it's flying along.

DR. LETELLIER: Or it's sliding on the floor.

CHAIRMAN WALLIS: And then, when it gets

to the blue region, it presumably doesn't instantly fall out. It sort of goes out and makes a pattern downstream, so you have --

DR. LETELLIER: There's an opportunity for a drift.

CHAIRMAN WALLIS: It's not clear there's nothing in the blue region. It's in the process of falling out there, but there may still be some in suspension.

DR. LETELLIER: That's certainly true, and we are more concerned at the moment about the suspended debris than the potential for sliding on the floor.

CHAIRMAN WALLIS: Where is the sump in this picture? Do you have that screen in this picture?

DR. LETELLIER: At the bottom of the annulus.

CHAIRMAN WALLIS: The very bottom of --

DR. LETELLIER: There's a bright spot.

DR. CHANG: It's sort of green in the center.

CHAIRMAN WALLIS: So it's enclosed by the red stuff.

DR. LETELLIER: Now, CFD models are one

methodology that the staff is familiar with for estimating the velocity counter. There are alternatives. The NEI is currently looking at open channel network flow models as an approximation to the bulk flow.

We are evaluating -- we will be evaluating that as an acceptable method. There is a potential for success. There is a wide range in the fidelity of the models. But in both cases, you have to make assumptions about how you're treating the variability in your input conditions. That's a common question that has to be addressed in both cases.

Again, the linear flume test characterized the incipient flow and settling velocities of our major debris types. And that database, in combination with velocity estimates, can be used to estimate transport fractions.

As far as the acceptable methods and what debris transport references are available, we've talked about using CFD versus network flow. Again, there are no integrated models specific for debris transport. Logic charts are the best systematic approach to assessing this fraction.

We do have peer reviewed articles on our CFD modeling of our scale tank tests that appear in

nuclear technology. But, again, they are very specific interests. They are limited in scope.

And, finally, the list of references on slide 32.

CHAIRMAN WALLIS: It wasn't clear to me that the CFD modeling was systematically compared with data from the tasks. It seemed to be qualitatively predicting the right sort of thing, but I didn't see a measure of how well it did quantitatively.

DR. LETELLIER: They were qualitatively compared using tracer objects to map the velocity zones, and --

CHAIRMAN WALLIS: But there isn't the quantitative verification or validation, or whatever you want to call it.

DR. LETELLIER: We felt that the pedigree of the codes for doing open channel flow was sufficient, given a qualitative comparison. We observed the same transport behavior of the fine debris as would be predicted by the velocity patterns.

MR. CARUSO: Did the people that did the CFD modeling know what the tests -- know the test results?

DR. LETELLIER: Of course. They were performed at the same time.

MR. CARUSO: Did they know the results of the test before they did the modeling? Was it a blind calculation, or was it an open calculation?

DR. LETELLIER: As a matter of protocol, the calculations and the tests were not conducted independently. But as a matter of practice, there were no initial conditions presupposed in the calculation that were defined by the test, except for the volume of water that was introduced and the geometry. I personally performed the calculations, and there was no intent to fine tune the calculations.

MR. CARUSO: I'm not asking about intent. I'm asking, did the people that did the calculations know the results of the experiments before they did the calculations?

DR. LETELLIER: The answer is no. The calculations were performed before the velocity mapping was done in the tank. And then, the qualitative comparison was performed. There wasn't a rigid protocol followed for blind assessment in that manner. But the calculations preceded the tests.

There are a number of references available that describe debris transport. The most current are listed as the NUREG-6882 in the middle, small scale tests for separate effects characterization, and then

also 6773, the integrated tank tests that incorporated rotational flows and a scaled geometry.

And at the bottom, I mention the peer reviewed articles that appear in Nuclear Technology.

DR. CHANG: Okay. The next slide is about sump screen head loss. When you have the collection of those debris at the sump screen, the next step is to consider the head loss.

In the Reg. Guide, we have the following positions. For the fully submerged sump screens, NPSH available should be determined from the conditions specified in the plant's licensing basis. But for the partially submerged sumps, both in Appendix A and also in Section C.1.3.4.4, we have the same statement.

That is, pump failure criteria should be assumed to occur when the head loss across the sump screen is greater than half of the submerged screen height or the NPSH margin. Either one, whichever is worse.

And then, estimates of head loss caused by debris blockage should be developed from empirical data. You have to have -- to do tests on those.

CHAIRMAN WALLIS: Do you see, though, what I mean about in the second bullet --

DR. CHANG: Yes.

CHAIRMAN WALLIS: -- you've got the screen, and it's behaving like a dam, and there's a loss there. And then, you've got the NPSH, and nothing -- isn't the loss across the screen -- doesn't that actually decrease the NPSH as well? It's not as if it's one thing or the other.

DR. LETELLIER: Calculations of NPSH generally start at the screen location. They account for the static head above the pump. They don't account for friction losses on flow paths preceding or prior to arrival at the sump. They do account for friction losses in the plumbing in the piping.

CHAIRMAN WALLIS: They do account for this loss through the screen, then, don't they?

DR. LETELLIER: The traditional definition of NPSH does not account for pressure loss, pressure drops, across the debris bed. That's being incorporated now as a point of comparison. If the pressure drop is greater than this failure criteria, then you will lose NPSH.

MEMBER RANSOM: Is it true, then, that you're just calculating the hydrostatic head available at the pump over and above the vapor pressure or the fluid?

DR. LETELLIER: Essentially, that's right,

with various regulatory arguments about credit for containment overpressure.

CHAIRMAN WALLIS: What's the argument again about this half of submersed screen height?

DR. LETELLIER: I need a diagram in order to illustrate. But you can imagine a vertical screen that's only partially submerged.

CHAIRMAN WALLIS: All right.

DR. LETELLIER: There's water on both sides of the screen, and debris is building on one side.

CHAIRMAN WALLIS: All right.

DR. LETELLIER: The pump is demanding a constant volumetric flow.

CHAIRMAN WALLIS: So there's a drop in level from one side to the other.

DR. LETELLIER: Yes, as the debris builds up.

CHAIRMAN WALLIS: Right.

DR. LETELLIER: But most importantly is if you cannot satisfy the volumetric flow, if there's not enough static head in the pool to force water through the bed, then you will -- your level will drop catastrophically, and you will lose NPSH.

The only pressure available to push water

through the bed is the static head of the pool. And on average, averaged across the bed, you have approximately one-half the --

CHAIRMAN WALLIS: Well, that means you'll simply suck the downstream part dry.

DR. LETELLIER: That's correct. As that level drops, you will lose NPSH by definition, because it's dominated by the static head above the pump inlet.

MEMBER RANSOM: What's magic about the one-half, though? Is that the limit of the pump's capability?

DR. LETELLIER: No. You have no mechanical advantage, because the pressure is equal on each side of the screen.

MEMBER RANSOM: No. But what I meant is, the pump only cares about what NPSH is available before it starts cavitating. So is the minimum NPSH, then, roughly half of the available head at the pump inlet?

DR. LETELLIER: No. The definition of NPSH from the point of view of cavitation is defined entirely separately.

MEMBER RANSOM: I know that. But why are you using the factor of a half?

DR. LETELLIER: Because there are two failure mechanisms. You can -- if you lose margin, you may cavitate, or you will cavitate at the pump. One realistic sequence for losing margin is a debris blockage that cannot satisfy the volumetric flow.

CHAIRMAN WALLIS: Even with the NPSH satisfied, the pump is working fine.

DR. LETELLIER: That's correct.

CHAIRMAN WALLIS: It just sucks all the water out, and it can't get back in.

DR. LETELLIER: That's correct.

MEMBER SIEBER: The NPSH disappeared.

DR. LETELLIER: And eventually you will lose NPSH, because your pump is --

CHAIRMAN WALLIS: You'll ingest air.

DR. LETELLIER: -- not running. You're ingesting water there. That's right. And so we're suggesting that you need to examine the minimum of these two criteria, both the NPSH -- because one leads to the other. If, for example, the one-half pool depth is less than the NPSH margin, if you have less than one -- if you have a pressure drop that exceeds one-half the submerged screen area, you will eventually lose margin. One precedes the other.

MR. ARCHITZEL: It's essentially

equivalent to the mid-loop operation in PWRs. When you get below mid-loop, you lose it.

MR. CARUSO: You have on page 6-2 of the -- do you have a copy of the --

DR. LETELLIER: Yes.

MR. CARUSO: -- knowledge base there? Page 6-2, you've got sump configurations. What sort of sump configuration are you talking about that would apply here? We all have copies of this. Page 6-2.

DR. LETELLIER: None of these figures actually show the water level. But if you look at E, the box type filter that has a vertical screen, the case that we're talking about is where the water level is only perhaps halfway. It has only submerged half of the screen, and the upper portion is open, so that you have containment pressure on both sides of the screen.

MR. HSIA: Bruce, can we go to --

CHAIRMAN WALLIS: And you can pump it out from the place faster than it can come in through the screen.

DR. LETELLIER: That's the motivation for the failure.

MR. HSIA: There's a better picture in the Reg. Guide.

DR. CHANG: Figure A.3. That shows the --

CHAIRMAN WALLIS: 3.6.1. is okay. E is okay, too. You can pump it out faster than it can get in by gravity through the screen.

DR. LETELLIER: That's correct.

MR. CARUSO: Do you allow them to calculate how it's going to build up and overflow as a function of time? I mean, there's always water pouring in, and gradually the water levels rise. Is that permitted?

DR. LETELLIER: I think that was the point of one of the comments that was made. And, in fact, we did --

DR. CHANG: Yes. As a function of time, we said, that you can consider this is -- right.

DR. LETELLIER: If they choose to do that.

CHAIRMAN WALLIS: So you might recover the pump again. I mean, as the water rises.

MR. CARUSO: As the water rises.

CHAIRMAN WALLIS: If it's not destroyed already.

MR. ARCHITZEL: But that was principally for like the plants that start spray very early and don't have that level yet. So they need to have a very -- it's not the full flow rate. It would be like

an ice condenser that's starting to spray early. The water is very low. They started right at -- initially. Not all the plants do that.

DR. LETELLIER: So we continue?

DR. CHANG: Bruce, yes.

DR. LETELLIER: So we continue with slide number 34.

CHAIRMAN WALLIS: It's the thin bed.

DR. LETELLIER: The final step of vulnerability assessment is head loss across the screen, given a presumed debris bed. And the head loss correlations that are -- it's shown generically below the figure, was developed for 6224 and validated against test -- experimental data for a limited combination of debris -- the predominant combinations of fiber, RMI, and particulate.

This figure shows the range of head loss on the vertical axis that would be incurred as a function of bed thickness, essentially fiber volume, for a given screen. There are assumptions here of velocity and area.

CHAIRMAN WALLIS: It's a very strange curve. You put in more fibers, you get less head loss.

DR. LETELLIER: One of the limitations of

the correlation is the assumption of a homogenized bed. And if you have a large fiber volume that's well mixed with particulate, it allows greater porosity, more flow area, and so the head loss is lower than if you have a very thin contiguous bed of fiber. The thin bed, one-eighth inch --

CHAIRMAN WALLIS: This is with a certain constant amount of particulates and more fiber dilutes than particulates? Is that the idea?

DR. LETELLIER: Each curve represents a different mass of particulate.

CHAIRMAN WALLIS: Okay. So more fiber dilutes the particulates.

DR. LETELLIER: That's correct. So you can see the reason for the minimum.

CHAIRMAN WALLIS: It shows that it isn't as simple as you think. And also, the compressibility -- the degree to which the pressure drop across the bed itself compresses the fibers has a big effect here.

DR. LETELLIER: And that's accounted for in the correlation.

CHAIRMAN WALLIS: Right. But if the fibers happen to be squishier than predicted, they can really clog up the --

DR. LETELLIER: That's correct. And that is a phenomena that we observe in some debris types. Calcium silicate, in particular, tends to reweld into a contiguous obstacle.

CHAIRMAN WALLIS: It concerned me a bit about other chemical products produced in the sump that will make this stuff gooier or whatever.

DR. LETELLIER: That's true. And if we have time to share the summary of chemical testing, you'll see that we are not confident that the 6224 correlation is robust for those debris types.

Of all of the steps of the accident sequence, I'd have to say that the head loss has been investigated in the most detail largely due to the amount of work that the industry did to actually design and test the strainer retrofits for the BWR resolution. There is a large body of information that became available at that time.

The head loss correlation has been implemented in a PC utility called BLOCKAGE. It's available for use by the public. It actually has some amount of verification and validation that's documented in the user's guide. It did not adhere to a formal software quality assurance plan, but they were conducted separately.

The code results were recorded separately in separate programming language, and then the results were verified. And then, the output from BLOCKAGE was exercised against all available test data to show the validation steps.

It's important that the head loss correlation be used with appropriate material properties. And, again, the database is not all inclusive. There are materials out there that have not been tested.

It's important that the -- that any alternative correlations be validated through comparable test procedures. The NRC work has established an expectation of quality and level of procedure and attention to detail that should be typical in any alternative method that's proposed.

Ultimately, if these head loss correlations are implemented to validate a new test -- or, I'm sorry, a new design of the strainer, then performance tests of these designs should be done comparable to what was done for the BWR resolution.

The head loss references on page 36 again mention 6224, which I wanted to remind you was actually issued as a draft NUREG. So it was subject to public comment, and the resolution is documented in

the appendix to 6224.

DR. CHANG: There are about 80 pages of the resolution of comments in this document, so it's extensively -- being extensively reviewed.

CHAIRMAN WALLIS: You don't consider the mechanism where the fibers sort tangle up on the screen? If anybody has cleaned out a drain from a shower, they noticed the hairs, though the screen is pretty coarse. It's very simple. It doesn't take all that many hairs to tangle up around there and block it up.

DR. LETELLIER: Yes, certainly. We have demonstrated that the thin bed can be established on screens as large as --

CHAIRMAN WALLIS: It's not just a bed. I mean, it can be actually something that goes around the -- extends downstream from the filter itself.

DR. LETELLIER: That's true. And it does -- you do incur some amount of head loss because of that, but the greater concern is a contiguous map.

CHAIRMAN WALLIS: Right. So are you finished now with your presentation?

DR. LETELLIER: So T.Y. has some closing --

CHAIRMAN WALLIS: Yes. My colleague Dr.

Rosen has to go in about five minutes.

MEMBER ROSEN: Yes.

CHAIRMAN WALLIS: Do you have something you would like to --

MEMBER ROSEN: Yes. Presumably, we'd go around the table at the end and get -- have some committee discussion. I beg your indulgence to just listen to my one comment, and then I have to go.

And that is that of all of the -- and I've stayed fairly close to this. I went to the PWR workshop in Baltimore. The committee agreed with me doing it, and I did do it, and you'll soon get my trip report.

But the thing that -- the only jarring thing I heard today that was new was that there is no plan by the industry to deal with -- in the guidance with material that goes through the sump subscreen, how one does -- what one does to analyze that. And that seems to me an open circuit in the protocol that's being developed.

We'll get to the very end of it, and then that question will be asked, and there will be no answer except -- I don't know what. Maybe John Butler of NEI or someone else could help me with that. But I guess I didn't really hear the answer to that.

That's my input.

CHAIRMAN WALLIS: When will we get your trip report?

MEMBER ROSEN: Well, it's done, and it ought to be -- I've given it to the staff.

CHAIRMAN WALLIS: It might help us with the letter that we're supposed to write to --

MEMBER ROSEN: Well, the trip report was rather, you know, brief and preliminary. So I'm not sure it will be much more than you heard here. I think you might just want to, you know, scan it.

MEMBER RANSOM: Who did you send it to?

MEMBER ROSEN: Sherrie.

CHAIRMAN WALLIS: Okay. Well, are you going to finish up?

DR. CHANG: Yes, the last one. In closure, I just want to describe the ongoing research activities under Generic Safety Issue 191. There will be a meeting before the end of October this year. We have two test reports coming out. One is the calcium silicate head loss test report. The other one is some very preliminary chemical tests done for the --

CHAIRMAN WALLIS: Are you going to test just the kind of chemicals that might be in the sump and at the temperatures that they might be at or --

are you going to test the paints and things that might be there?

DR. LETELLIER: We can go through the --

CHAIRMAN WALLIS: So we can get on to some of these questions about --

DR. CHANG: Yes. Bruce has some slides here.

DR. LETELLIER: If you'd like to go through the summary, there's better information.

DR. CHANG: And long term is up to end of fiscal year '04. We plan to have a debris sample characterization of PWRs. There we tried to collect sample latent debris from five volunteer plants, and then we tried to do some additional head loss tests on them. And we plan to have HPSI throttle valve clogging study as well. And the --

CHAIRMAN WALLIS: Can you do something about this zone of influence issue that seems to be of some concern? And if there's anything you can do from what's already happened in Barseback, and so on, to see, was it a directional jet, or was a spherical thing, or anything that would help to give some realism to the zone of influence model, that would really help I think. I'm suggesting that you do that.

DR. CHANG: I don't know if Barseback --

they still have all this information available or not.

CHAIRMAN WALLIS: This report lists a lot of things that are described, but then someone should go in and say "ah ha." Now, from what I saw, what the description is, this shows that it is a jet or something -- something you could deduce from it that helps your model.

DR. LETELLIER: Those aspects can be revisited. I rather doubt that there will be any --

CHAIRMAN WALLIS: But try.

DR. LETELLIER: -- new inspiration that comes forth.

CHAIRMAN WALLIS: But you have an ongoing contract, do you? You can do this?

DR. LETELLIER: I'd be happy to do it. I just need some direction.

MEMBER SIEBER: Well, did the Ontario hydrotesting tell you anything? That was actual configurations with varying types of insulation.

DR. LETELLIER: Those were free jet expansion for two-phase flow. They were very similar to the air surrogates that were performed for the BWR study.

MEMBER SIEBER: So they don't tell you much about energy distribution in a compartment.

CHAIRMAN WALLIS: Didn't the University of New Mexico -- they did a test where they had a pipe with insulation on it, and they took a two-phase jet and directed it at it?

DR. LETELLIER: That was actually done as part of the BWR.

CHAIRMAN WALLIS: But that's not a spherical jet. That's a directional jet.

DR. LETELLIER: That's correct. And it was done for the purpose of measuring the damage pressure, so that we know what the vulnerabilities of each insulation type are.

CHAIRMAN WALLIS: Yes. But then, you didn't go back and say, "Now, if we had assumed it was a spherical jet, what would we have got for the predicted pressure."

DR. LETELLIER: Had we done that, we would have been taking credit for dissipation that we didn't show.

CHAIRMAN WALLIS: Yes. It seems to me that you're doing a test which is at odds with your whole model for zone of influence.

DR. LETELLIER: I don't quite understand the comment. We are remapping the equivalent damage volume into a sphere as a practical simplification, in

light of the fact that our targets are distributed rather homogeneously throughout containment.

CHAIRMAN WALLIS: Well, I guess we'll revisit this again. I'll get back to the firehose and the -- you know, the firehose -- the purpose of the firehose is to go in one place. It's very different from the spherical.

DR. CHANG: Bruce, do you want to go into your second subject?

DR. LETELLIER: Yes.

DR. CHANG: Bruce would like to share with us some of his ideas about other alternatives.

DR. LETELLIER: Let me ask the committee what your preference would be and your time constraints. We are --

CHAIRMAN WALLIS: Tell us what's important in a short time. We don't have to go -- I don't think my colleagues have to go exactly at 12:00, so we can go at, say, 12:30 or so.

DR. LETELLIER: We're here at your convenience. We have two topics that --

CHAIRMAN WALLIS: Well, if you had 20 minutes, could you tell us the most interesting stuff here?

DR. LETELLIER: My personal opinion is

that the chemical effects testing is the most interesting.

CHAIRMAN WALLIS: Okay.

DR. LETELLIER: I think you can browse through the set of handouts on sump operability strategies and see what conceptual concepts that you could put on a table in a brainstorming context. The fact that we're presenting these does not imply any endorsement, practicality, or operability of these concepts.

But we hope to show you that the industry is not without options. There are a number of things that can be pursued through --

CHAIRMAN WALLIS: We can look at these pictures, and we can sort of see how they might work.

DR. LETELLIER: Exactly. That's the intent.

CHAIRMAN WALLIS: So now you want to tell us about the chemical --

DR. LETELLIER: Give me a moment to pull up the slides. And I apologize, I don't have handouts for you. Those can be provided after the briefing.

MR. ARCHITZEL: I've got an ADAMS number if you want it. It's already in ADAMS. It's the same slides you did at the --

DR. LETELLIER: It's a condensed set. This is essentially the information that was presented at the workshop two weeks ago.

MR. ARCHITZEL: Workshop slides are available in ADAMS.

DR. LETELLIER: Have members of the committee looked at those slides --

CHAIRMAN WALLIS: No.

DR. LETELLIER: -- already?

CHAIRMAN WALLIS: No, I didn't.

MR. HSIA: I think it would be more appropriate -- when we are complete with the chemical testing, there will be a report issued, and at that time we can come back, if you choose, to present to you a complete picture.

CHAIRMAN WALLIS: But the fact that you're saying there are chemical reactions that produce precipitants indicates to me that there are chemical reactions which produce significant stuff. And it may not always be in the form of nice, dry -- dry-type particulate stuff. It may be gooey or bubbly or something, depending on what's going on chemically.

DR. LETELLIER: That's correct. And we can show you the status of our investigations. We've been looking at this for the past three or four months

over the summer.

We had a rather limited scope to assess the plausibility of these various chemical reactions, exacerbating head loss on the screen. All of this work is also being done at the UNM Civil Engineering Department in their hydraulics lab.

The motivation for the work -- you're well aware of concern of the committee regarding gelatinous material observed in TMI. Try to focus on determining where this material came from and if it's a plausible concern for reactor accident sequences.

CHAIRMAN WALLIS: Now, this stuff that sprays out from the reactor is borated water. So when it hits the stuff out there in the containment, it's boric acid. When it gets in the pool, it gets diluted with sodium hydroxide. Is that the way I understand it? It gets turned to a high pH in the pool.

DR. LETELLIER: Your chemical injection tanks that actually increase the pH -- there are sodium hydroxide injection tanks, and the boron concentration in the RWST is much different than is present in the RCS.

CHAIRMAN WALLIS: But they don't inject into the reactor system. They inject into the containment somewhere?

DR. LETELLIER: Yes, they do. They are part of the spray and also the --

CHAIRMAN WALLIS: But they don't inject into the RCS. The RCS is borated, acidic, low pH stuff. So the jet that hits the walls is acidic, and then it has a chance to do its acidic reactions. It runs down the wall, and it meets this alkaline stuff, which has come from somewhere else, and the spray, which has fallen down from the roof.

MEMBER SIEBER: The reaction actually occurs in the containment atmosphere.

CHAIRMAN WALLIS: It can as well. I think it's -- there's some uncertainty. But the jet -- if the jet's direction is certainly acidic, and it sputters acid all over the wall --

DR. LETELLIER: That is a detail that we have not examined as yet -- the time dependence of the concentrations. We've looked at more the homogenized solution of the containment pool, what would be -- in particular, how the spray RWST impacts the water chemistry, because keep in mind that the sprays impinge on a much larger area of exposed metal than the break jet.

I should state right up front that this is important enough -- we feel it's a very important

issue, and we've convened a peer review panel to take place the first week of September. We're not sure what expectations we should meet through this peer review, but we do have outside experts that are -- both represent academia, national laboratories, and the industry, that are not currently participating in the safety -- resolution of the safety issue.

We're investigating several tasks. The scope was broad -- look at chemical effects. We focused primarily on the corrosion of exposed metals with subsequent precipitation. There are other chemical effects. One has been postulated this morning -- hydrogen generation that leads to the formation of bubbles. That was not on our list of priorities.

CHAIRMAN WALLIS: But it's mentioned in your report.

DR. LETELLIER: As I said, it's not on our list of priorities for this limited introductory effort. We were looking at chemical degradation of an existing fiber bed. We're concerned about --

CHAIRMAN WALLIS: Well, the simplest thing you can do -- I guess I could ask the staff here to do it -- is it says here that it's already being used to estimate hydrogen source term. So we could simply

take the results of what -- that analysis that has been done, find out how much hydrogen there is.

DR. LETELLIER: That's true. It could be done as an analytic exercise.

CHAIRMAN WALLIS: Find out what that information is. If it's -- it says it's been done.

DR. LETELLIER: I will tell you that none of our immersion samples show evidence of bubble formation. Zinc granules, zinc coupons, paint chips, the generation is not --

CHAIRMAN WALLIS: So they're used in many FSARs to estimate hydrogen source term.

DR. LETELLIER: Yes.

CHAIRMAN WALLIS: So they must be there somewhere. Okay.

DR. LETELLIER: That's true. And because of that work, we do have estimates of exposed aluminum area, because of that need to track hydrogen generation. We have some idea of plant vulnerability regarding exposure area.

MR. ARCHITZEL: I just asked the committee a question on that. If a lot of the hydrogen generation comes off the severe accident source term in the DBA, would that be a factor that we should consider?

CHAIRMAN WALLIS: No.

MR. ARCHITZEL: Well, I mean, you assume a certain amount of fuel damage. It's much more than you get in the DBA.

CHAIRMAN WALLIS: Is that what's done in the FSAR?

MR. ARCHITZEL: Hydrogen generation is coming from the radiolytic decomposition of the water. And if the DBA prevents that, I know we assume that in the DBA. The whole purpose of this exercise -- you've got a DBA, and you've got this -- I mean, there's an opportunity to not consider it based on the fact that you don't -- I know we do consider it for radiological doses.

MEMBER KRESS: This thing is to keep from getting these products in there, and I don't --

DR. LETELLIER: Exactly.

MEMBER KRESS: -- think you want to do that here.

CHAIRMAN WALLIS: There's a different question, though. There's --

DR. LETELLIER: Hydrogen generation does not precede loss of sump.

MEMBER KRESS: No, that's right.

DR. LETELLIER: It's not an initial

condition from our --

MEMBER KRESS: That's right. I don't think you want to do it that way.

MR. ARCHITZEL: I don't understand. I guess I'm just trying to -- I guess my point was there is an opportunity to not consider it because you haven't had the core damage.

CHAIRMAN WALLIS: No, I don't think you can not consider it. It's a chemical effect. I mean, it says it here -- that the aluminum reacts with --

MR. ARCHITZEL: Well, the aluminum -- okay. I thought it --

CHAIRMAN WALLIS: -- the stuff that's in the sump, and it's going to make -- so it makes hydrogen. So I think you have to consider it.

MR. ARCHITZEL: Okay.

DR. LETELLIER: We will review that.

CHAIRMAN WALLIS: Maybe you can't use the numbers for some other calculation to find out how much hydrogen. So maybe what I'm asking Ralph to do is not very helpful. But at least he can look at it and see if we can learn from it.

DR. LETELLIER: A similar issue brings to mind the water chemistry. In a severe fuel damage event, you'll have nitric acid produced because of

radiolytic exposure of air.

MEMBER KRESS: Still, you have to have the radiation.

DR. LETELLIER: You have evolution of chlorides from cable trays. But, again, those conditions --

MEMBER KRESS: That doesn't seem to be part of this.

CHAIRMAN WALLIS: We're not looking at radiation here.

MEMBER KRESS: Right.

MEMBER SIEBER: They're longer term.

DR. LETELLIER: Those are longer term, and we've ignored those chemical reactants in our matrix. Schematically, this is the concern that occurs, that the borated solution and sprays impinge on exposed metal surfaces. Metal surfaces are also immersed in the pool, and the metals can be dissolved and suspended as free ions in solution.

If you reach saturation -- and these metals are extremely insoluble -- once you reach saturation, there's a potential for precipitation to occur. The formation of metallic hydroxides, for example, shown in the middle panel -- we've confirmed that, yes, you can produce these effects using

background, borated water, and simulated metallic nitrates to induce free metal, dissolved metal. You will generate a precipitant, and it will cause significant head loss.

CHAIRMAN WALLIS: When these zinc ions are running around in this low pipe pH -- now it's high pH, isn't it?

DR. LETELLIER: Yes, it is.

MEMBER KRESS: Aren't these extremely small?

CHAIRMAN WALLIS: Is there a hydrogen production mechanism in there?

DR. LETELLIER: I'm not sure. We have not addressed --

CHAIRMAN WALLIS: But it says in your report that's why --

DR. LETELLIER: Well, I don't have any comment. We have not addressed hydrogen generation.

CHAIRMAN WALLIS: Let's see if there is.

MR. CARUSO: What comes out from the reactor, the spray or -- to create the zinc hydroxide? It's --

DR. LETELLIER: In the middle panel, that's the precipitation.

MR. CARUSO: Actually, it's the left-hand

panel, the production of the zinc hydroxide I believe generates hydrogen.

DR. LETELLIER: That may be true. We have not examined that.

CHAIRMAN WALLIS: Well, where does the H go from the water? It presumably gave the OH to the --

MR. CARUSO: That's where it goes.

DR. LETELLIER: Well, keep in mind that this is strongly buffered solution. There's a lot of sodium hydroxide that's available.

CHAIRMAN WALLIS: You're going to sort that out.

DR. LETELLIER: Right.

MEMBER FORD: I guess the thing that -- everything you've said so far is thermal dynamically possible. I'd question the kinetics of the process, whether it's a big deal or not, and --

DR. LETELLIER: That's a very important issue. We have demonstrated that each of these separate effects can occur.

MEMBER FORD: Yes.

DR. LETELLIER: We've demonstrated the linkage between step 2 and step 3. The flocculent material is very transportable. The particulates are

extremely fine. They are extremely small, but they also are hydrophilic in the sense that they bind water molecules into a jelly, into a gelatinous mass.

CHAIRMAN WALLIS: What you need to do in your picture is show those green things as zinc and the white things as hydrogen.

MEMBER KRESS: That's what they are.

CHAIRMAN WALLIS: And then, if you take a physical model, the zinc particles and hydrogen bubbles -- and your picture is very good --

DR. LETELLIER: We add the hydrogen bubbles.

MR. CARUSO: Does the concrete have any chemical effect?

DR. LETELLIER: It certainly does. In fact, we added some calcium carbonate to represent the eroded concrete present in the jet.

MR. CARUSO: And what did it do?

DR. LETELLIER: Well, we have not done an exhaustive assessment of the parameters of concentration. In general, it increase the pH similar to the sodium. And, in fact, in that respect it's a buffer solution, and it increases the solubility of the metals, dissolved metals.

MR. CARUSO: So it probably also depends

on the type of concrete, too.

DR. LETELLIER: It certainly would. Containment environments are very dirty. You have a variety of chemicals that are different from our clean test tube, our beaker experiments.

The executive summary we have already hinted at. Metal corrosion is credible for the borated cooling water. The UNM test confirmed the literature reported values for room temperature. They are typically reported in units of grams per hour per square meter of exposed metal.

But for the elevated temperature, we did oven tests at 80 degrees C, just to represent a substantially higher temperature. We were not able to confirm the reported rates of 11.3 grams per hour per square meter. Those are extremely high rates, and we believe that the kinetics are an important aspect of this inconclusive test at the moment.

We are looking at immersion and corrosion in a quiescent beaker that's not subject to flow mechanisms of any kind. And we think that there's a surface catalyzed redeposition of this material. It's not freely released to the solution so that you gradually reach saturation. It's affected by the very high local concentrations near the substrate, and I'll

show you a picture of what that corrosion product looks like.

CHAIRMAN WALLIS: Now, in terms of breaking off flakes of paint, does the boron in the water help to loosen up the flakes?

DR. LETELLIER: We haven't examined that issue. We are looking at paint as a debris source that's liberated in the zone of influence. The industry guidance and the best available NRC guidance is to assume 100 percent destruction of paints within the damage zone. We have not looked in depth at the chemical effects on those paint chips.

We are concerned about the potential to leach the zinc from inorganic primers, because that's a significant reservoir of metal, for example. And we do have some very preliminary tests, qualitative in nature, that I wasn't prepared to present.

The second bullet in blue -- we have confirmed that the low solubility of these metals does lead to precipitation if you exceed the saturation threshold, and that the chemical precipitate does lead to a substantial head loss in combination with fiber on the screen.

Ultimately, the plant vulnerability depends on the connection between corrosion and

precipitation, and that really needs to be established by an integrated test that we haven't done yet.

To get into the details of how the test procedure was conducted --

CHAIRMAN WALLIS: Well, let me ask you here. Suppose that these chemical effects are important. We don't seem to have a knowledge base for industry to respond to the question about, do they have a significant effect on their plant? So what's required to get that knowledge base? I would presume NEI isn't going to create it out of nowhere. It's not there.

DR. LETELLIER: We're in the process of creating the knowledge base, and our October report will be the first --

CHAIRMAN WALLIS: So, again, it's one of these things where you can't do the analysis, because you don't know yet. So we don't do anything.

DR. LETELLIER: Well, I'm not sure that's true. I mean, I showed you a corrosion rate that's reported in the literature. It's very high.

CHAIRMAN WALLIS: It is.

DR. LETELLIER: It will lead to many hundreds of pounds.

CHAIRMAN WALLIS: Orders of magnitude

bigger than you observe.

DR. LETELLIER: Yes. But, in fact, that's the best available evidence at this moment. And the conservative approach is to adopt that corrosion rate with some estimate of exposure opportunity, and assume the connection between corrosion and precipitation.

MEMBER FORD: In your knowledge-based report, this one here, you mention in one of the -- some of the plants that have seen blockages. Sludge was talked about, in which there was metal corrosion plugs. What metal was it?

DR. LETELLIER: Well, in the BWRs that have a suppression pool, the sludge is predominantly iron oxide. It's rust. And, in fact, during the BWR study, that debris source dominated considerations of additional dust that might be present, because there is so much iron oxide.

And the correlations were tailored to perform best with that type of debris. There were no considerations of the chemical precipitants at that time.

Let's skip to some information about how the precipitation tests were conducted. We started with the ionized water supplemented with our boron concentration, representative of actual plant

configurations. It was not done in the detail that you alluded to. It was meant to describe the gross -- the bulk mixed concentration.

We established a fiber bed in a closed loop test apparatus, and then we introduced a metallic salt in order to force the precipitation to happen. We essentially introduced more -- we introduced enough metal to exceed the saturation threshold, and then we observed the results.

MR. CARUSO: Why did you pick those particular chemicals?

DR. LETELLIER: We were mostly interested in the metals, because there are exposure opportunities for iron, aluminum, and galvanized cable trays represent zinc. We could also have used copper, lead, etcetera. Those are the vulnerable materials in containment.

We use the nitrate as a convenience for introducing dissolved metal. We could prepare a stock solution, essentially.

CHAIRMAN WALLIS: Did you have -- you didn't have flakes of zinc paint and stuff in there, but --

DR. LETELLIER: Not in these tests. That's correct. These are some of the test samples

that are arrived at from different concentrations of iron and zinc. In the next slide, I'll show you the data that all of these metals --

CHAIRMAN WALLIS: We are looking at little buttons of stuff which got eaten by --

DR. LETELLIER: We're actually looking at the test samples about four inches in diameter.

CHAIRMAN WALLIS: They obviously got eaten by something. They got corroded, didn't they? Isn't that what we're seeing?

DR. LETELLIER: No. Let me clarify.

CHAIRMAN WALLIS: It looks like artifacts from an archaeological dig or something.

DR. LETELLIER: They look like jellyfish. Keep in mind that we introduced clean fiber into a test section that's about four inches in diameter. We put in 100 grams of fiber, which is essentially yellow insulation. And to this test column we introduced the metallic nitrate, induced the precipitation, and we measured the head loss.

CHAIRMAN WALLIS: So you made zinc precipitate on the fiber in some --

DR. LETELLIER: Yes, we forced it to precipitate.

CHAIRMAN WALLIS: And you made this

gooey --

DR. LETELLIER: Yes. These are the test samples of the bed that was recovered from each test.

CHAIRMAN WALLIS: -- cookie dough type stuff. Is that right?

DR. LETELLIER: Right.

CHAIRMAN WALLIS: Well, that's the sort of thing we thought might happen, or could, you know? Needs to be considered.

DR. LETELLIER: We can certainly create those conditions in a confined environment. It is plausible.

These are the data, head loss being shown on the vertical axis, and the effective concentration of each metal along the abscissa. The blue line shows you the baseline. That's the head loss incurred by the fiber alone, by itself. And then you can compare the margin that was measured.

CHAIRMAN WALLIS: It's just like the precipitates from the soap that gum up your drain. It's not just the hairs. It's the other things that get in with them.

MEMBER SIEBER: That holds the hair together.

DR. LETELLIER: That's right. And that's

an important factor. You should understand that the head loss that we're observing here is much, much greater than you would observe from an equivalent amount of dry particulate. It's much different than the 6224 correlation.

The threshold of about 10^{-3} molar is the threshold for the saturation threshold. That's where we first start to observe the effects. The concentration axis really represents additional mass that we've added to the bed, and that's why the pressure -- the head loss trends are consistent between materials.

MEMBER KRESS: That's the concentration you would have had if you added that mass and none of it precipitated?

DR. LETELLIER: Yes.

MEMBER KRESS: Okay.

DR. LETELLIER: That's right. But, in fact, all of it precipitated.

MEMBER KRESS: Yes.

DR. LETELLIER: And all of it arrives on the bed.

Now, just to give you some engineering chemistry facts to kind of baseline your understanding about this, first of all, you understand that every

metal has a different molecular weight. But if you look at this block, the threshold for precipitation of about 10^{-4} molar is equivalent to several tens of pounds in a million gallons of water. That's not very much material. And, in fact, most large drives don't have a million gallons.

CHAIRMAN WALLIS: The amount of hydrogen needed to float that stuff, because hydrogen is so light, is even less.

DR. LETELLIER: Perhaps you're right.

CHAIRMAN WALLIS: It doesn't take many models to --

DR. LETELLIER: Keep in mind that the precipitation -- the precipitant -- it might serve as a nucleation site for bubbles.

MEMBER KRESS: And now that was a log scale on concentration.

DR. LETELLIER: Yes.

MEMBER KRESS: And so you're going up a factor of 10. Do you have that much available compared to the amount of water you have?

DR. LETELLIER: Well, keep in mind, your observation was exactly correct. These are the concentrations that would exist if there was no precipitation. But, in fact, once you reach the

threshold, it does precipitate.

MEMBER KRESS: So it's a continuous --

DR. LETELLIER: Yes, it is a continuous process. And so this concentration really represents the amount of material that we force on the bed. It's directly proportional to the mass on the bed.

MEMBER KRESS: Okay.

DR. LETELLIER: Now, in this block, at 10^{-3} molar, the amount of material that we actually added to our 10 liter closed loop is very small -- fractions of a gram -- .3 grams of aluminum were added to this test volume.

And we are actually inducing seven to 10 feet of head loss with just a fraction of a gram. That's much, much different than you would expect from an equivalent mass of dry particulate.

I did not bring the electron micrographs of the debris bed. But you can see that the precipitant tends to stick or adhere to individual fibers, and it changes the hydraulic flow characteristics of the bed.

Dry particulate, by contrast, tends to lodge in the interstitial space and obstruct the flow area. So there's a quite different mechanism going on.

Also, our observations of the jelly-type layer give you the impression that it's taking up a much larger volume than would be assumed by those fractions -- fractions of mass. So the precipitant is actually hydrophilic.

CHAIRMAN WALLIS: That's what gels do.

DR. LETELLIER: Exactly. It binds water molecules into a gelatinous mass.

CHAIRMAN WALLIS: Which really makes me feel good.

MR. CARUSO: I'll tell you where there's a lot of information about this, and that's in the filtration industry. And I'm -- probably every plant in the United States, every nuclear plant in the United States, has a chemical waste treatment building that has a whole bunch of filters in it with pre-codes, and all sorts of techniques like that to do exactly what you're trying to measure. And the people that sell those machines know all about how this works.

DR. LETELLIER: That's exactly right. For the final steps of water quality treatment, for clarification they add an aluminum nitrate coagulate.

MR. CARUSO: Flocculents.

DR. LETELLIER: Exactly.

MR. CARUSO: And somewhere in every plant there is a chemical engineer that runs that waste treatment plant that knows all about this chemistry. You've just got to get that guy out and talk to the thermal hydraulicist.

DR. LETELLIER: But keep in mind this is the kind of chemistry that you do not want inside a containment during accident. So there is a disconnect in the application of their expertise. But you're right; there's a large body of information available.

MEMBER SIEBER: It's the same process, though.

MR. CARUSO: The same process.

DR. LETELLIER: Yes.

CHAIRMAN WALLIS: Well, I think you've convinced at least me that this is something that needs to be considered in resolving this issue of some --

MEMBER SIEBER: It may go beyond that. This may be the overarching consideration.

DR. LETELLIER: Let me introduce one more observation from the tests that are not so conclusive. When we tried to confirm the dissolution rates at high temperature, we assumed that corrosion would happen in a more or less uniform manner until you reached that

saturation, and then the precipitate would form.

But, in fact, we never reached saturation. We started to produce a secondary corrosion product that recrystallized on the metallic substrate. And this may be an artifact of our very quiescent beaker where, in fact, you cannot remove the dissolved metal that's free to enter the solution. You're dominated by local concentration effects.

But, in fact, the corrosion is evident at high temperature. Shiny zinc granules turn black, and they tend to gain mass, in effect, leading us to suspect that there's a secondary chemical reaction that's binding either nitrogen from the air, dissolved air, carbon from the air, oxygen, something -- and we're working to analyze the composition.

CHAIRMAN WALLIS: When they oxidize in this solution, they take the oxygen from something, presumably.

MEMBER KRESS: OH.

DR. LETELLIER: From the water.

CHAIRMAN WALLIS: What's left behind?

MEMBER KRESS: H_2.

DR. LETELLIER: These corrosion products have a very interesting crystalline structure. There are a couple of different formations. The very -- the

small fine platelets that are well organized crystal structure, and there's an alternative which is the large puff balls, for lack of a better word.

The compositions of those two crystals are very similar. We've done an electron X-ray spectrum analysis as a byproduct of the electron micrograph. You get an excitation signature from electron shells, and you can look at the X-ray spectrum and identify composition. And we've done some of that.

I didn't choose to present it, but it's helping us understand the composition in hopes that we'll pin down the formation -- the mechanism for formation.

CHAIRMAN WALLIS: Is this the picture of the black stuff on the tiny zinc particles?

DR. LETELLIER: Yes.

CHAIRMAN WALLIS: So the size of this is actually still small compared with the particle itself. The size of these --

DR. LETELLIER: The scale of the white bar at the top is 20 microns.

CHAIRMAN WALLIS: But the size of these growths --

DR. LETELLIER: Very small.

CHAIRMAN WALLIS: -- barnacles and all,

they're very much smaller than the size of the big particle itself.

DR. LETELLIER: Yes.

CHAIRMAN WALLIS: So it's not as if it grows very big as a result of this.

DR. LETELLIER: That's correct. And, unfortunately, this material is quite frangible. It comes off. It's brittle, and it -- depending on further testing, it may represent a new debris source.

CHAIRMAN WALLIS: Another concern, you know, might be if it enabled the particles to hook up together or something, if you stick them together, make some other structure.

DR. LETELLIER: Perhaps.

CHAIRMAN WALLIS: Anyway, very interesting.

DR. LETELLIER: So our status to date -- we've essentially completed all of the experiments that we had proposed under the current scope, and now we're documenting our results that will be released as a NUREG in the October timeframe.

There are significant uncertainties related to corrosion at high temperature. We have two hypotheses that -- either the dissolution is happening so quickly that you reach saturation and immediately

deposit into crystals, or you're dominated by surface chemistry. You have a heterogenous reaction occurring that's dominated by the local concentration.

MEMBER KRESS: The 11 grams per hour per unit area, does that come from extrapolating the erroneous curve from lower -- a lower temperature?

DR. LETELLIER: I suspect that it might, and that's the reason that we felt it --

MEMBER KRESS: You can miss those pretty much, depending on how much of the bottom part of the curve you have.

DR. LETELLIER: We felt it necessary to confirm those rates before we proceeded with our assessment of vulnerability.

MEMBER KRESS: Yes. But you're quite right. These local effects could -- local concentrations could have a big effect.

DR. LETELLIER: And I think we could do a better job of this measurement, corrosion rate measurement, if we had a flowing system.

MEMBER KRESS: Yes. Stick a stirrer in your beaker.

DR. LETELLIER: Yes.

MEMBER FORD: The other thing is that you're using zinc as a -- correlated to zinc chromate.

Although some paints do have metallic zinc in them, not many do. It's not zinc chromate. There's an inhibitor, and so you're merely fooling yourselves by doing your experiments on metallic zinc.

But, you know, it would --

DR. LETELLIER: We're also concerned about the galvanized cable trays, which represents an additional source of zinc.

MEMBER FORD: That would be metallic, too, although not entirely. If it's a more modern plant, it wouldn't be zinc, it would be zinc granules.

CHAIRMAN WALLIS: So is there potential here for this thing -- this issue to be sort of resolved by you sending out all of this -- this Reg. Guide, and NEI comes up with a wonderful analysis, and everyone says everything is fine.

And then, in a year or two's time, people have done a little more work with this chemistry and have said, "No, it isn't," because the chemical stuff is much more lethal to the screen than all these fibers or in combination with them. Therefore, you've got to start again. Is there a potential for something like that to happen?

MR. ARCHITZEL: I'll say that industry said no way. They're not going to go to their VP and

do this with this issue hanging out there. That was what got --

CHAIRMAN WALLIS: So nothing is going to happen until the chemical issues are resolved?

MR. ARCHITZEL: That was the feedback we got at the workshop.

CHAIRMAN WALLIS: That's what we heard from I think Steve Rosen. He said that because you can't understand what's going on, you do nothing.

MR. ARCHITZEL: I think that was the comment he made, yes.

MR. HSIA: But at this moment, I would like to put a different perspective -- yes, it's true based on the tests we have done so far that there is significant head loss because of the gelatinous material. What we really don't know is how much metal structures or metal parts that could interact with the coolant at lower part of containment.

Even the spray comes on -- there are metals up there. We really don't know how long -- what the effect is. We know the corrosion will be there, but we don't know whether the corrosion will be carried down and start to react. So there are still a lot of questions.

We're not saying at this moment that

plants have problems. All we're saying is if you have gelatinous material. So it's very plant-specific.

DR. LETELLIER: I'd like to point out also in the handout package on sump operability strategies, on the second-to-the-last page, there is a concept of sacrificial screen area, which might be appropriate to mitigating this problem.

CHAIRMAN WALLIS: Yes. There are all kinds of fixes one might devise when one understands enough about what's happening.

DR. LETELLIER: Including chemical balance on the lines of phosphate baskets that were introduced for iodine sequestration.

CHAIRMAN WALLIS: Right. As long as you don't screw up something else by doing that.

DR. LETELLIER: It has to be an integrated safety --

CHAIRMAN WALLIS: I think we're coming to the end here. My colleague Dr. Kress has to leave. I'd like to ask him to give us the benefit of his thoughts at this time.

MEMBER KRESS: Okay. First off, I do think this is a significant safety issue, and I'm glad to see all of the good technical work that's been done so far.

I'm a bit surprised that there is no element of risk-informing this Reg. Guide or risk-informed the rule associated with it. And what I mean by that is I think one could attach probabilities to breaks of different sizes.

And if one had, then, an acceptable frequency of these breaks based on the outcome -- and the outcome would probably be as a release of fission products or something in the long-term cooling -- then one might be able to -- if one had an acceptance value on that, one might be able to eliminate many of the break size based on risk considerations, and get down to a size that may be a reasonable size for screens that we may already have.

So I'm a bit surprised that I don't see that thinking showing up so far. And along the same vein, I think leak before break would be an input in establishing these frequencies. And I'm surprised not to see that.

Another thought is I -- you know, in spite of the comments on reflection off of surfaces, the zone of influence still looks to me like it could use some more thought. I would have guessed, for example, one might have taken the conical shape and just directed it arbitrarily in all different directions

and pick out the direction that gave you the most, or something like that.

And I still think it needs some more thought, and I haven't gelled my own thoughts on that.

And finally, I think this life stuff you showed on the chemistry has the potential to be a real showstopper. And I think eventually need to put to rest chemical effects.

Now, I suspect the kinetics may be too slow for this to be real significant, and so I think it's real important that you get through the kinetic effects and actually pinpoint what the potential danger in that is. But anyway, I think you guys are thinking along the right ways, and looking at very important phenomenon. And I'm glad to see some good technical input going into it.

CHAIRMAN WALLIS: Well, that's good, Tom. How about, then, this Reg. Guide -- how does that fit in from the regulatory point of view? Is it going along with all of these -- thinking along the right lines?

MEMBER KRESS: You know, I feel a little bit like Steve. I hate to see nothing being done. And the question is, you know, Reg. Guides are usually living documents. You change them, as you learn more

and learn more. The question is: when do you stop learning and put out something that's useful?

I don't know. That's a regulatory decision, and I don't know if I've got much advice there. But personally, I don't think the Reg. Guide is quite far enough along to be ready to go out. But, you know, I think we need to look at it more and look at the NEI document in combination before we can make that decision.

CHAIRMAN WALLIS: Thank you.

It's a bit of a chicken and egg situation, isn't it? I mean, you send out the Reg. Guide and ask for all kinds of things, and this may induce people to do the work. Or you can say, well, they're going to adapt the Reg. Guide. We want to see what work they can do before we fit in the Reg. Guide, so the Reg. Guide fits in with it. And you have different strategies that could be adopted there.

MEMBER KRESS: Thank you, guys. I have to run.

CHAIRMAN WALLIS: Let's see. Tony, do you want to wrap things up, or T.Y., or anyone from the staff?

MR. HSIA: The only thing I want to say is thanks for this opportunity. You have pointed us to

several important issues we may not have delved in deep enough. We're going to do that, and that's it. I don't have any other concluding comments.

CHAIRMAN WALLIS: Thank you.

MR. HSIA: Thank you.

MR. BUTLER: I feel compelled to make some clarifying statements on what industry is doing to address this issue. I do not want to leave the ACRS with an impression that the industry is not doing anything to address the issue awaiting final resolution on the chemical effects. We are doing a number of actions, what we can do right now with the information we have.

We just completed a workshop. We are doing -- individual plants are doing walkdowns to assess their inventory of possible debris sources, to address their layout, to get as much information as they can, such that when they're given a go-ahead to do the evaluation, they can do that.

The concern expressed at the workshop mentioned by Tony earlier was that the final resolution, the final fix, it would be very difficult to go to a VP right now and say, "We need to install a 600 square foot passive screen" without knowing the effect of the -- of that solution -- of the chemical

effects on that solution.

So without having a little bit more information, we -- you know, we're -- the final resolution may be delayed until we have that information. So what we're going to do is meet with the staff on -- in September to discuss what research, whether it be NRC sponsored or industry sponsored, is necessary to get the answers as quickly as possible.

CHAIRMAN WALLIS: Will you have the September -- the guidance you were going to put out ready in September, with this chemical issue as something to be done later? Or what?

MR. BUTLER: We're hoping to get that out as -- end of September, maybe a little bit later than that. Whether or not we have the chemical effects addressed in that completely or --

CHAIRMAN WALLIS: Okay. So we'll --

MR. BUTLER: -- as just a placekeeper --

CHAIRMAN WALLIS: We'll have something to look at, then.

MR. BUTLER: -- we'll have something to look at, yes.

CHAIRMAN WALLIS: As far as the physical effects.

MR. BUTLER: Yes. Again, we're not trying

to hold this up. In fact, we're trying to speed the resolution up as much as we can, because this is a costly item just to keep following this. So resolution is sought by all parties.

Thank you.

CHAIRMAN WALLIS: Thank you.

So I think I will thank you, presenters, from the staff and from Los Alamos. And I'll turn to my colleagues, yes, for their input.

Do you want to start, Peter?

MEMBER FORD: Sure.

CHAIRMAN WALLIS: Are you ready to go?

MEMBER FORD: I thought the Reg. Guide and the associated materials that were given to me, they recognize all of the constituent parameters in the sequence of events leading up to sump blockage and loss of NPSH.

I think we recognize all the relevant ones. The only question, of course, is chemical and precipitation. And I agree with Tom; I think that the kinetics of the process may well assure that it is not a major one. It has to be tested.

It gives good advice as to how to tackle the analysis of the various specific effects, individual effects, in the debris source and

transport, etcetera. My concern is that there is no quantification of the integrated effects between those various parameters.

And the validation of that quantification against what was observed at various plants -- and those plants are itemized in this knowledge base report. And, therefore, I can't see how the licensee can demonstrate that they can avoid the failure criteria that is given in Appendix A.3 or the Reg. Guide.

The reality is, however, that it will take I think a fair amount of work by the licensees, NEI, EPRI, or whoever it is, to demonstrate that they can meet those criteria in A.3.

I'll be very interested to see what NEI comes up with in September as guidance to their customers. I think that the Reg. Guide should be issued now in its current form, with the proviso that work is done by the industry to resolve these outstanding questions.

I don't know how that is done, procedure-wise or procedure process. But I think it is a safety issue, and it should be -- we can't just wait forever for these outstanding questions to be answered.

CHAIRMAN WALLIS: Thank you.

Jack?

MEMBER SIEBER: Okay. I'll be brief. This is Rev. 3 of this Reg. Guide, and I am certain there's going to be a Rev. 4, because I don't think that this represents a complete investigation of all of the effects that are important in the sump blockage issue.

I don't know, but my feeling is the chemical effects is an important phenomenon. And I've done some work, but I'm struck by the fact that I think that it may be the overriding effect that's based on some simple things that I've seen in my career.

And I think it's important when you do it that you actually, instead of looking, for example, at elemental zinc that you test based on the compounds that you will find in containment, so that you get the right reaction instead of saying, "Well, I tried sodium hydroxide and a coupon of zinc, and I didn't get this," or "it took this long to do it." I would rather see you use zinc chromate and actual galvanized coupons of the same stuff that's in the plant as opposed to trying to simplify the experiment. And so I think that that's an important factor.

So the question becomes -- do you issue

Rev. 3 now, or do you say, gee, we don't know enough about everything that's important; why don't we learn everything that we can, and then issue some final Reg. Guide? And I guess I come down thinking that what's in the Reg. Guide is not incorrect, even though there are some assumptions that folks can question.

But it's not incorrect. It may be incomplete, but I think the industry knows that, and the staff knows that. So when I ponder whether or not it should be issued or not, I guess I come down on the side that it ought to be, with the expectation that research has to continue, and that there will be a further revision.

And I don't think that you can resolve with certainty whether plants comply with the three or four general design criteria or not until you know a little bit more about these effects. So that would be my opinion.

CHAIRMAN WALLIS: Vic?

MEMBER RANSOM: Well, I guess I'd like to support what Dr. Kress suggested, that there seemed to be many opportunities for risk-informing this sort of thing, and as opposed to an Appendix K conservative-type approach that's being taken.

It also turns out, coming from the west

and all the irrigation ditches that are around and the paper mills and -- as well as sewage plants, I can't believe that you'd want to rule out or be careful not to rule out solutions which include active trash mitigating schemes as well as inactive ones.

I mean, with a system where you can essentially eliminate the problem, I don't know how that factors into a plant. That's another issue.

But, and a lot of those schemes, too, would eliminate I think the chemical aspect of the problem, if it exists. So whatever is put in the Reg. Guide I think should allow the freedom to employ these kinds of things, if they desire them. So --

CHAIRMAN WALLIS: Well, I think I've already said most of the things I would say in summary. I think I agree with what I've heard from my colleagues. It seems to me it's a question of regulatory strategy. We've put out this Guide saying that all of these things need to be considered, and then say wait for industry to respond.

My expectation is that they will not be able to respond very well. And then, the question is up, really, to the NRR folks -- what do you do? What do you do in a situation like this where it does have an effect on safety, where there are -- there's even

-- there's chemical questions which no one really knows the answer to yet, which may turn out to be quite significant in terms of the answers they give.

So it's a very interesting example of a regulatory situation where some kind of wisdom is called for on the part of people administering the regulations. And that's really where I think we need some good answers, because you can put out the Reg. Guide as it is and say, yes, it's not perfect, it's a living document, but at least it gets things going.

And then we can say industry is going to respond. There's also the actions that NRR is taking. It's being played out. I, for one, will be very interested to see how it does play out, and I can't, though, see a sure route to a happy conclusion for everybody.

MEMBER SIEBER: Well, it's sort of interesting -- you know, Section B, which is in every Reg. Guide, is implementation. It sets forth the situations where the Reg. Guide will be used, and it's pretty limited here. I think there's three of them. You know, it's 50.59 things, but it doesn't take the form of some generic communications or a bulletin or anything like that that tells a licensee, "You go out and reexamine your sump." Maybe that comes later.

That's another step in the process.

And so I think industry, unless some issue comes up that forces them into this Reg. Guide, they could sit back until such time as NRR decides -- or the Commissioners decide -- you know, we want everybody to demonstrate compliance. And they could do that at any time, and, in fact, an inspector in the plant could do that. He could ask for the licensee's calculations.

MR. ARCHITZEL: But imposing the Reg. Guide would be a backfit. They would have to go through CRGR, if it's on more than one --

MEMBER SIEBER: That's right. And that's why D is written the way it is, I presume, because the first one talks about new construction, plants that aren't built yet. The second one is application of 50.59 and, you know -- and so I can see the strategy just from the words that were used.

MR. ARCHITZEL: That's 50.59 comment actually comes from 1985 --

MEMBER SIEBER: Yes.

MR. ARCHITZEL: -- where the issue was it wasn't cost beneficial. But as you do 50.59 changes, consider that in terms of when you're placing out insulation. But in point of --

MEMBER SIEBER: It comes with an extra factor.

MR. ARCHITZEL: Right. It's a little different. It's estranged in the Reg. Guide. But the other point is if the committee considers it appropriate to examine -- I mean, Research is going before the CRGR. You could change this into a potential -- it's a lot different. You could do a cost-benefit..

MEMBER SIEBER: Yes.

MR. ARCHITZEL: It could be considered differently before the CRGR.

MEMBER SIEBER: Yes. So, anyway, to me the strategy is sort of obvious as to what it is you're doing. That's okay. You know, that's the way regulation works. That's the way this agency does things, and I don't see anything wrong with it.

CHAIRMAN WALLIS: Okay.

MR. HSIA: Let me just put in a couple of new pieces of information. The meeting with CRGR is August 26th, and so -- and the meeting -- we are coming back to the full committee on September 11th, and we are meeting with industry on September 10th.

So things are going to happen. Decisions will be made and recommendations will be made by a lot

of different people. So we will keep -- fully keep your staff informed. Therefore, you will be informed.

And I'm guessing that you will make a decision on this Reg. Guide after the full committee, is that correct?

CHAIRMAN WALLIS: Yes. No decisions are made except by the full committee.

MR. HSIA: Okay. So by that time, we'll wait and see, see whether there are other inputs. Maybe it can make your decision a little easier. But in any case, it's not an easy one. We realize that. And we thank you for giving us the time.

CHAIRMAN WALLIS: Okay. With that, I'd like to close the meeting, and I will do so. We are now adjourned.

(Whereupon, at 12:39 p.m., the proceedings in the foregoing matter were adjourned.)

Official Transcript of Proceedings

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS) 505th MEETING, DAY 2

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THURSDAY,

SEPTEMBER 11, 2003

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ROCKVILLE, MARYLAND

The Committee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Dr. Mario V. Bonaca, Chairman, presiding.

COMMITTEE MEMBERS:

MARIO V. BONACA, Chairman
GEORGE E. APOSTOLAKIS, Member
THOMAS S. KRESS, Member
GRAHAM M. LEITCH, Member
DANA A. POWERS, Member
VICTOR H RANSOM, Member
STEPHEN L. ROSEN, Member
WILLIAM J. SHACK, Member
JOHN D. SIEBER, Member
GRAHAM B. WALLIS, Member

ACRS STAFF PRESENT:

JOHN T. LARKINS, Director
SHER BAHADUR, Associate Director
SAM DURAISWAMY, Technical Assistant
B.P. JAIN
HOWARD J. LARSON, Special Assistant

ALSO PRESENT:

BRUCE BEISLER, Florida Power and Light
T.Y. CHANG
KEVIN COYNE
MARY DROUIN
NOEL DUDLEY
MICHELE EVANS
JOHN FLACK
STEVE HALE
DONNIE HARRISON
TONY HSIA
MICHAEL JOHNSON
JOHN KAUFFMAN
PT KUO
ERIC LEEDS
BRUCE LETELLIER
TILDA LIU
RON L. LLOYD

ALSO PRESENT: (CONT.)

LEDYARD (TAD) MARSH

MIKE MAYFIELD

JIM MEDOFF

GARETH PARRY

MOHAMMED SHUAIBI

HOWARD VANDERMOLEN

A-G-E-N-D-A

from the Subcommittee meetings. We do value the comments that you've got and hope we can end up at the right place. So thank you very much.

CHAIRMAN BONACA: Thank you. Okay, the next item on the agenda is draft final revision 3 to Regulatory Guide 1.82 Water Sources for Long-Term Recirculation Cooling Following a LOCA. And Dr. Wallace will take us through this presentation.

MEMBER WALLIS: This is an interesting and important issue for almost 30 years. It's been revived at various times when various events occurred which changed people's view of what might happen. It was tackled for the BWRs, and after a lot of activity in the 1990s the owner's group got together, the staff made it clear what had to be done. And all the BWRs changed by sump screens. Sometimes by making a large area of change in the sump screen.

We have recent work at Los Alamos which showed pretty clearly that there was an issue for PWRs. And so we're here to hear what the staff is doing in terms of a regulatory guide to resolve this issue.

This doesn't put to rest the TSI, which is associated with this problem. And we have both the staff and Los Alamos here today. I'd like to ask Mike

Mayfield to get us started.

MR. MAYFIELD: Thank you. We're here this afternoon to present to you and seek Committee endorsement of the publication of the final revision three to Regulatory Guide 1.82. We met with the Committee when we had the draft to put out for comment. We've been out. Gotten the comments. Have addressed those comments. And we believe that we have addressed them in such a way that we're ready to go final with the guide.

This is important for us to move forward on because it is, first of all, and important issue. But secondly, the staff has put out a bulletin to have licensees take certain actions. And to some degree, the licensees are looking towards this draft regulatory guide to provide guidance on how to address the bulletin, or at least as they begin to structure their responses.

In response to the public comments, we did make some changes to the guide that we believe are important to have on the street in the final form so that licensees are dealing with the staff's latest thinking, as opposed to the draft that was put out for comment. So we are hoping to get the Committee's endorsement so that we can move forward and publish

this document.

NEI is preparing guidance that's more detailed than what you'll find in this regulatory guide. The staff will review that guidance, and we have yet to -- we and NRR will review that guidance document once NEI has it. And the decision will be made at that time, what vehicle to use to endorse that guidance, assuming that that's the direction we go.

But in the interim, we felt like it was important to finalize this guide and get it on the street. I have with me this afternoon Michele Evans, who is the chief of the Engineering Research Applications Branch in Research, and Michael Johnson, who is the deputy director of DSSA in NRR. Tony Hsia and his team will make the presentation on the guide and answer your questions.

MEMBER WALLIS: Mike, I forget exactly what words you used about the guide, but you're viewed to say it was going to get the utilities going and responding to this issue.

Now, if you read the guide, it seems to me it very clearly tries to cover all the gamut of phenomena which are likely to happen which influence all these events. But it doesn't say much at all about what's an acceptable way to analyze those

phenomena. Many guides go further in terms of saying we'll accept this method, that method, or something.

And I think the Committee's going to ask you about whether the methods for analyzing these phenomena are available, and how good they are.

MR. MAYFIELD: Okay.

MEMBER WALLIS: Because that's not really tackled in the guide at all.

MR. MAYFIELD: That's correct, it is not tackled in the guide. There is some technical background information. And I think perhaps the best thing I can do is let Tony and his team try to address that.

MEMBER WALLIS: Well, I want to say at the outset, I think it's going to be one of the questions we have.

MR. MAYFIELD: I understand.

MEMBER ROSEN: Right around that question also, I'd like to ask the question of have you seen the draft NEI guide? Is there such a thing that you've looked at?

MR. MAYFIELD: I have not. Bruce is shaking his head yes. So perhaps they have. I think it is a fair statement that we have not officially reviewed and taken a position on that guidance.

DR. LETELLIER: That is correct. We don't have an official position. But we've been interviewing interim appendices of this draft. We have not viewed it in its integrated whole.

MEMBER ROSEN: So there is some work that you've already looked at, and it's moving.

DR. LETELLIER: It is mmoving.

MEMBER ROSEN: Okay. That's good.

DR. LETELLIER: And they are still committed to their September deadline, I believe.

MEMBER WALLIS: So let's proceed now. Tony?

MR. HSIA: My name is Tony Hsia. I'm the Assistant Branch Chief in ERAB in Research. Thanks for this opportunity to be in front of you and present to you our Regulatory Guide 1.82 revision three.

To my right is Bruce Letellier, our contractor from Los Alamos National Lab. To his right is Dr. T.Y. Chang, staff with the ERAB in Research. What we plan to do this afternoon is I'll go over the overview and the background of this issue which some of you are very familiar with. Then I'll turn over to T.Y. He will continue to go into more detail of the Reg Guide. And if any other technical details, both T.Y. and Bruce will be able to pick that up.

At the outset, I would like to say this Reg Guide is the same as any other Reg Guide. We may not have said specifically we will accept this model, we will accept that model. But by definition we do say in the beginning of this Reg Guide say Reg Guide will describe acceptable methods to the staff in evaluating your vulnerability to the debris impact on the sump performance.

MEMBER WALLIS: Perhaps it's what we mean by "methods" that's at stake here. I mean, it says you must consider debris formation, debris transport, and all that, but it doesn't say what methods you use to consider those things.

MR. HSIA: Correct. This Reg Guide is not a prescriptive Reg Guide that lays out the methods in detail because as you all know this issue is an extended issue for many years. We have many, many NUREG reports in there that are much more detailed are described in there.

So I believe during the Subcommittee briefing we did attempt to refer to those references. But this afternoon we'll try to address those specific questions also.

If I may have viewgraph number 2. Okay, this is the structure of this afternoon's briefing.

I'll cover the background, the reasons for issuing this Reg Guide, and the use of the Reg Guide, and Reg Guide 1.82 activities associated with Revision 3 of this Reg Guide. And then T.Y. will pick up with the remaining of the presentation this afternoon.

Viewgraph 3. As you know, this issue started almost 30 years ago when Revision 0 was issued in June of 1974. At that time, the whole industry as well as us knew little about the impact of debris on the sump. So the best thing we could do at that time was make a conservative assumption. So we assumed 50 percent blockage of the sump screen. And when you calculate the net positive suction head for your recirculation flow.

And then after that, we realized we need to do better. We start to cnduct research, and also the NRC issued USI-A 43 in January of '79. That USI is focused on containment emergency sump performance.

Shortly after that, Revision 1 of this Reg Guide was issued that provided guidance. The guidance was based on USI-A 43 resolution. In early 1990s, several nuclear power plants, starting with Barsebaeck in Sweden, and then followed by several BWRs in this country, including Perry, Limerick, Grand Gulf, and Browns Ferry had experienced suction strain or

blockage events that in some cases demonstrated the recirculation flow was negatively impacted because of the blockage of the sump screen.

And we realize we need to do more. We need to have more knowledge. Therefore, more research was conducted starting at that time. We issued Revision 2 in 1996. That was a revised guidance with the focus on BWRs.

Also, NRC issued Bulletin 96-03. That's to specifically focus on the potential plugging of strainers and BWRs. And that bulleting requested licensees to implement measures to ensure ECCS functions following a loss of coolant accident.

And also for that revision, instead of using the old 50 percent blockage, we recommended that the licensee during their evaluation to assume 100 percent debris transport from the break location to the sump. That's a conservative assumption. Unless they can justify otherwise. Again, that's a conservative assumption.

Come to this point. Today we're ready to present to you and seek your endorsement of Revision 3. This Reg Guide, like Mike said earlier, and our colleagues at NRR would like to use this also as a guidance toward contributing to the resolution of GSI-

191. That is a BWR sump performance. Next viewgraph, please.

The reason for issuing this Reg Guide, as I said earlier, is to contribute to the resolution of GSI-191, and also to provide an enhanced debris blockage evaluation guidance for PWRs and methods that's acceptable to the staff.

As all Reg Guides, I said earlier, they are not substitutes for regulations. Therefore, compliance is not required. But those are the acceptable methods to the staff for evaluation of the debris impact on sump performance.

Of course the other methods the licensee would like to propose we certainly will consider, and will review individually for acceptance at that time. Viewgraph 5.

Earlier this year, in February, we came in front of the ACRS, briefed the ACRS. At that time it was DG-1107. That also included with NRR presentation on GSI-191, also their plans for the generic letter. At this moment, I understand the generic letter is planned to be, the draft is to be going out for public comment toward the end of this year. And the final generic letter is expected spring of next year.

Back in, I believe in June or earlier,

there was a Bulletin 2003-01 issued by NRR. That bulletin requested the licensees to either demonstrate they satisfied the requirements in 50.46 on long-term cooling, or they had to take an interim compensatory measure to ensure ECCS performance.

I understand that we have received responses from licensees on that bulletin. The majority of them chose to use compensatory measures.

So the public comments on this version of Reg Guide was received after April of this year. We have addressed those public comments. And T.Y. will discuss all of that in more detail later.

And that will bring us to today. As we said earlier, we did brief the Subcommittee in August, and we have gone to CRGR, also in August. And that leads us to where we are today. T.Y.?

DR. CHANG: My name is T.Y. Chang, Office of Research. Slide number 6. There are a lot of key revisions in this version of the Reg Guide. The majority of the modifications of this revision was focused on the pressurized water reactor section in order to enhance guidance on how to evaluate debris blockage issue.

And we tried to utilize the information from the prior Revision 2 version for the boilers.

Wherever applicable, we tried to use those information. And also, in addition, we added inside scan from the research and the GSI-191.

After the revision of the PWR sections, then we turn our attention to BWR sections as well, trying to make sure that the two sections are consistent to each other. Also, in the BWR sections, we also added the staff's position on the evaluation of BWR owner's groups URG. That's a Utility Resolution Guidance for the ECCS suction strainer blockage. That's for the PWR plants.

Finally, within this version of the Reg Guide, another Reg Guide is subsumed into this one. That is Reg Guide 1.1, the net positive suction head for ECCS and containment heat removal system pumps. So Reg Guide 1.1 will no longer be in existence. It will be part of Appendix A of this Reg Guide.

MEMBER LEITCH: Some of this work, as I understand it, is based on recent testing that was done. Recent test results at Los Alamos, was it?

DR. CHANG: Yes.

MEMBER LEITCH: My question really is does any of that test data invalidate the work that was done on BWRs?

DR. CHANG: Maybe Bruce?

DR. LETELLIER: Not that we're aware of. There haven't been any apparent contradictions at this time. In fact, much of the guidance is based on the same guidance that was issued for the BWRs, as far as methodology.

MEMBER LEITCH: But this recent test data was done after the changes were made to the BWR suction screens.

DR. LETELLIER: That's correct. I think the focus of the research program under GSI-191 was to increase the depth of the database on debris transport properties.

And also we had hoped to do some two-phase debris generation tests because that was not part of the BWR study. We had more success on the transport and head loss characterization than we have on the two-phase debris generation.

But we were focused on the unique aspects of the PWRs, and so none of the research that's come to light has contradicted those earlier results.

MEMBER WALLIS: Well, I wonder if that's true. I mean, I've been reading your reports. There are many statements of this type, about larger quantities of fibrous debris could reach the strainers.

That being predicted by models and analysis, this is from the Barsebaeck event, that being predicted and methods being developed for resolution of USI A-43.

And then when you're talking about the presents state, you say preliminary findings suggest two phase jets can inflict significant damage at distances much further away than those measured either in USI A-83 studies or BWR earned-impact test program.

There are lots of statements like this in your document. Now, if the new tests show that things can happen further away and more bigger effects and all that than predicted before, this would seem to have some effect on the BWRs too.

DR. LETELLIER: Of course it would. And there are statements to that effect, that they need to be applied with full understanding of that phenomenology and adjusted appropriately.

And we tried to provide, in every case, examples of how to do that scaling where it was appropriate. The first citation that you quoted, the difference between the initial debris generation in the three-zone cone model, that was actually addressed by the BWR work.

And if additional conservatism and test

data were provided to cover that.

MEMBER WALLIS: So they did provide additional conservatisms?

DR. LETELLIER: Certainly.

MEMBER WALLIS: So it might be expected that the PWR would do the same thing?

DR. LETELLIER: That's our hope, yes.

MEMBER WALLIS: Okay.

DR. LETELLIER: But the recent bulletin was just to PWRs, not to all the science.

MEMBER WALLIS: Correct.

DR. LETELLIER: Okay.

DR. CHANG: The next slide is about the resolution of the public comments. The draft Reg Guide that was called DG 1107 was issued in February of this year, and there's a two-month period for the public to send in their comments.

And up to about 90 comments were received from seven commentors, including four utilities: Westinghouse, NEI and the one individual. In descending order of number of comments received, here is a list of the most raised comments.

The first one is a comment about a conformance issue for current plans. Our response is that this Reg Guide is generic in nature, and it may

go beyond current designs.

The intent is that this Reg Guide will be useful for future plans as well.

MEMBER WALLIS: Was that the issue that they raised? I thought the issue was --

DR. CHANG: The issue is that some of the other conformance --

MEMBER WALLIS: They will find themselves out of conformance if they do the analysis. What are they expected to do?

DR. Chang: This is -- most of the comments is that the current plan designs, in certain cases, are different from what's described in the Reg Guide. For instance, I think we mentioned that it's -- people should have two sumps in the PWR plant.

And some of the plants, they don't have two sumps. So, this is just to state the staff's position and give out acceptable methods to treat this ECCS problem.

Then, the next most asked issue is about--

CHAIRMAN BONACA: Now, just a question on that.

DR. CHANG: Yes.

CHAIRMAN BONACA: This is a Reg Guide, so this provides a means of addressing the issue. But

when you say that they should have two sumps, that's prescriptive.

I mean, it's not an option immediately, so what would be the approach for those plants that don't have two sumps. They'll have to make modifications, I guess, to --

DR. CHANG: Well, the Reg Guide, it is not a requirement.

CHAIRMAN BONACA: Right.

DR. CHANG: This is not a regulation.

CHAIRMAN BONACA: Yes.

DR. CHANG: So it just simply states the staff's position, and also the acceptable methods. Anything different than that is okay, if --

CHAIRMAN BONACA: I guess what I'm talking about is that -- I mean, if you establish some functional requirement of some type, then you can suggest ways to fulfill that requirement, to meet it.

And then you can leave it to the licensee to meet that requirement however he can do it. But if you prescribe two sumps, I mean that's not --

DR. CHANG: The intent is for the future plants.

CHAIRMAN BONACA: Okay.

DR. CHANG: It's desirable to have two

independent sumps.

MEMBER WALLIS: Is that only for future plants?

DR. CHANG: Pardon?

MEMBER WALLIS: Those are conformance issues for current plants. I mean, that's the whole question, isn't it? If they do this analysis based on the guide, they may well find they can't meet the long-term cooling criteria. What are they supposed to do then?

MR. HSIA: The real test -- the real test is whether you do have enough water to be fed into the reactor system during long-term cooling. The ultimate test is your net positive suction head.

Whether you have one or two or three sumps, if you can demonstrate -- let's say I only have one, but I can demonstrate what debris --

CHAIRMAN BONACA: Okay.

MR. HSIA: I can still meet the net positive suction head, then I'm establishing that I have no problem.

CHAIRMAN BONACA: So you're establishing a functional demand?

MR. HSIA: Yes.

CHAIRMAN BONACA: And you're suggesting a

way in which it can be done? All right.

DR. CHANG: And also, it's a function of the size of the screens, and so forth. There are a lot of different parameters you have to look into.

CHAIRMAN BONACA: Yes.

MR. HSIA: One of the complications of this issue for these B's or P's, is particularly for the P's, is very much plant-specific. And as a matter of fact, BWRs are simpler, because they are designed -- they are more or les similar.

And P's could have very different design compartments and so on.

MR. MAYFIELD: Mr. Chairman, this is Mike Mayfield. When you look at, under regulatory positions 1.1, the first sentence says ECCS stumps, which are the source of water, and so on, should contain an appropriate combination of the following features and capabilities.

And then the notion of having two sumps is one of those. It's not a mandate that you have to have two sumps.

CHAIRMAN BONACA: It's a way to fulfill --

MR. MAYFIELD: It's one way. And again, there's a fairly lengthy list of those kinds of things that would be desirable features. And you're looking

for some combination, so that you don't lose net positive suction.

CHAIRMAN BONACA: Sure.

MEMBER WALLIS: I still think the issue here was the plants anticipated, as a result of this, they would have to make changes. Even though you claim that no backfit is implied, they probably will, just as the BWR's made all these changes.

So there will be a lot of conformance issues for the current plants.

DR. CHANG: This issue came up in the CRGR discussion, the briefing we had with them, and we -- our position is that this is a conformance type of a backfit.

MEMBER WALLIS: Right.

MR. MAYFIELD: It's a compliance backfit.

MEMBER WALLIS: I think our overview of this is problem is that probably all the PWRs, as the BWRs, will make changes in the plant - most likely as a result of this issue being resolved.

MR. MAYFIELD: That could be an outcome.

MR. HSIA: In my opinion, it's really hard to say. It depends on the evaluation.

MR. MAYFIELD: Again, Doctor Wallis, I wouldn't want to presume that they're all going to

have to make changes. But the notion is that it could -- your statement could be an outcome of licensees evaluating this.

The BWR licensees evaluating their ECCS systems, that's possible.

DR. LETELLIER: I would further add that if changes are necessary, they will likely be in compliance with the Reg Guide. One before the other. If their individual vulnerability assessment warrants, they will make improvements along these guidelines.

MR. HSIA: As well as the coming NEI guidance -- industry guidance, so…

DR. CHANG: I don't know -- should I go on with --

MEMBER WALLIS: I'm not sure you need to go through all of these comments.

DR. CHANG: Okay, I can -- some of them -- some of the comments raised, I discuss them in the later slides as well.

MEMBER WALLIS: Yes.

DR. LETELLIER: Could you just discuss -- clarify what is meant by leak before break for debris source? I'm not quite sure what that means.

DR. CHANG: Well, this is the position that we responded to from a Westinghouse letter, we

stated that position. What it means is that the leak before break is not applicable when you try to consider how many amount of the re-generation can be created from pipe break.

So, for the purpose of estimating the amount of debris generation, the leak before break criteria cannot be used. This is in line with the 10 CFR 50.46 position.

That section is on the ECCS cooling. There, it says, in order to calculate the function of an ECCS, you have potentially many different locations of break, and try to find the most severe pace in order to design your ECCS system.

So this is in line with what is the position in the 10 CFR 50.46.

DR. LETELLIER: So, when you're looking at debris generation, you have to consider the instantaneous guillotine break of the largest pipe? Is that correct?

In other words, you cannot assume that there's a leak and you detect the leak and are able to shut it down. In other words, you have to assume that the line breaks and the debris is going to be generated as a result of that.

DR. CHANG: Well, people are considering

the double-ended guillotine break, middle sized break LOCA or small sized LOCA. But I think the position of the staff is leak before break is not acceptable for this purpose.

MR. HSIA: If I may jump in, the current agency position is leak before break and it can only be used for certain specific applications, such as pipe whip.

MR. MAYFIELD: This is Mike Mayfield. The change that we made to GDC 4, which is the one that deals with the pipe whip restraints and jet impingement barriers.

That allowed the elimination of those. The notion was that that change was adequate for eliminating the dynamic effects associated with such pipe breaks.

Then you get tied up with was this the dynamic effect or not. And my contention is that this is not a dynamic effect, this is an impingement effect.

And the notion of instantaneous double-ended, the notion is that you've got a jet that's potentially moving around. One of the other things to keep in mind is the leak before break size crack that we'll talk about for GDC 4, and that's been analyzed

as people have sought relief from having pipe whip restraints and impingement barriers.

That's a big hole in the side of the pipe. This is not weeping water. We had briefed the Committee several years ago. We would be happy to come back in and show you what that really means.

This is a significant leak. It is a -- in the large pipe, it is a very big hole in the side of the pipe. And, analytically, you'd have to move that around the pipe's circumference, to make sure you've captured the appropriate potential debris source.

So it actually complicates the analysis. Would it reduce the amount of debris generated? I think that almost certainly the answer to that is yes. Now you're left with, okay what's the trade-off.

The view that we've had is that one, you're hard put to really argue this is a dynamic effect. To include it at this stage would significantly -- would cause us to have to go back and revisit things that are in 50.46 and the change we made to GDC 4.

And we, at this stage, we were having some difficulty justifying making those changes for this specific application. My understanding is that the industry is making some overtures and pursuing that

line of discussion.

It's a policy issue that we'll be happy to entertain. But to move forward at this time, with this guide, we felt it was more appropriate to move forward, making the assumption of the double-ended break and deal with the debris generation on that basis.

MEMBER WALLIS: Can we move on?

DR. CHANG: Yes, the next slide, number eight. Here's a summary of Reg Guide 1.82, in terms of accident sequences. When a LOCA happens, the initial shockwave and blowdown jets impinging on the insulations will create the most amount of debris.

That usually happens in the first minute or so. So, we, in this Reg Guide, we are going to talk about our position, how we are going to partially the break location and what kind of sources should be looked at as a debris potential source.

And once you have those debris generated, in order to estimate how much of the debris will end up at the sump screen, the next step is to do the debris transport analysis.

That includes three types of transport. First is airborne debris transport. Right after the pipe break and blowdown, the air velocity in the

contaminant could reach 300 feet per second, according to some of the analysis.

So it's a very fast velocity within the turbulent situation in the contaminant. And the debris can be blown to the dome area of the contaminant. So this is the airborne debris transportation.

Of course, eventually most of it will settle down and come down. The next is after the --

MEMBER WALLIS: So, this 300 feet per second, do you have an idea what a stagnation pressure is for that?

DR. CHANG: I just read in the report that 200-300 feet per second velocity can be expected.

DR. LETELLIER: He's saying the displacement velocity, as the fluid stayed in -

MEMBER WALLIS: I'm saying that as a debris model for your Figure A-2, that says that after you get to a seven or something, the stagnation pressure's only half the psi.

It seems to me that 300 feet per second is a bigger stagnation, and you say it's all over the whole containment. That doesn't seem to be consistent.

MR. HSIA: Excuse me, I missed -- what figure are you referring to, Doctor Wallis?

MEMBER WALLIS: Figure A-2, the somewhat

notorious Figure A-2. It says that there isn't a L over D number on there, I think it's about seven is down to a half a psi.

I just brought that up because I think there are a lot of inconsistencies about this zone of influence on the velocities and the pressures that need to be sorted out. So, please go on…

DR. CHANG: Yes, this figure actually is a carryover from the A-43 document. We didn't put down the L over D numbers in the regions one, two and three here. But the --

MEMBER WALLIS: They are in your report. And I can see that seven is the L over D number that's out --

DR. CHANG: Yes, this is just a conception to show that --

MEMBER WALLIS: Well, this is not a conception, this comes from work done by Sandia.

DR. LETELLIER: No, the intent of the figure in the Reg Guide is conceptual.

MEMBER WALLIS: Yes, but the figure in the -- now, come on, this is an exact copy of the figure that's in the basis.

DR. CHANG: We deleted the L over D numbers there, within the three regions.

DR. LETELLIER: It's intended to show the --

MEMBER WALLIS: You see the problem I have, is that I look at this, I see that everything gets exhausted by a certain distance. And then here's someone telling me that I've got velocities in the whole containment, which are bigger than I see from this figure.

You know, that's at a much lower distance. That's why I brought this up, that's all. Let's move on.

DR. CHANG: Later on, Bruce has some view graphs to talk about the ZOI, so -

MEMBER WALLIS: No, I want to talk about ZOI too.

DR. CHANG: We can go into that later on.

MEMBER WALLIS: Okay, so lets move on. Can we get the next slide?

DR. CHANG: Okay, then it's washed down. After the containments sprayed and then the debris was sent up at the basement of the containment and get washed, some of them --

MEMBER WALLIS: Okay, so it says here that ZOI can be used. The zone of influence is the zone in which the destruction occurs, right?

DR. CHANG: That's correct.

MEMBER WALLIS: And if I look at this figure I mentioned, I see that it says that after about five L over D's, there's limited damage. And then in another report from Los Alamos, the parametric study, it says that it's able to use a 12 diameter sphere.

Now, there's a different number, all right? And in other places I hear that the zone of influence, in oral presentations, can be as big as a third or half of the whole containment.

This just doesn't seem consistent with this figure which says that everything gets tired after about five L over D's.

DR. LETELLIER: This figure is intended to be conceptual, and I've suggested that --

MEMBER WALLIS: It's not, it's a guidance. I mean, it refers to -- this is conceptual in the guide, but if you look in the guide that you've put out as the technical basis, which I think is the basis suggested for use in all of these analysis, it has numbers on it.

DR. LETELLIER: This is the knowledge base you're referring to --

MEMBER WALLIS: If I pick and choose in

these knowledge bases, I can get a lot of different numbers.

MR. MARSHALL: Excuse me, my name is Michael Marshall, I'm a former project manager for this project. One reason those numbers vary is based on the type of insulation.

So, I think that's one reason why they probably removed the numbers from the graph. The larger one's for, let's say, an encapsulated fiberglass would carry out to that 30 or that larger L over D.

MEMBER WALLIS: 30 L over D?

MR. MARSHALL: Yes, a larger distance. Your metallic insulation, depending on the type of clap, again you get --

MEMBER WALLIS: Well, I agree with that. I agree with that. I agree with all of that. It's just that if I look at different parts of these reports, I sometimes see five, I sometimes see 12, I can even see 60 in one of these parts of the report.

And therefore, there's a great variability here. And, you know, it seems to me that different people can pick different numbers and use them in their analysis.

DR. LETELLIER: They can pick numbers and

use them inappropriately, certainly. The knowledge base presents a variety of models that provides a survey of historical development for the problem.

And Michael raises a very important point, that the damage pressure's very specific to the insulation type, so the damage pressure distances will vary according to what your targets of interest are.

And it's important that the licensees understand that.

MEMBER WALLIS: Oh, we know that. We know that. But --

DR. LETELLIER: The use of these figures, and I should apologize for borrowing old graphics, but they are intended to be conceptual, and I've recommended that --

MEMBER WALLIS: They can't be conceptual if they're going to be used in analysis. You've got to put numbers in.

CHAIRMAN BONACA: But, I mean, do you think that it's clear to a licensee, for example, based on the guidance you provide in the Reg Guide and the supporting information, if he would understand what numbers to use for what material?

DR. LETELLIER: There are supporting documents that recommend damage pressures for specific

insulation types.

MR. HSIA: If I may read, Bruce, the section in the current Reg Guide that refers to the figure you're pointing to. And I'll quote…

CHAIRMAN BONACA: What page are you at?

MR. HSIA: I'm at page 1.8-2.

CHAIRMAN BONACA: Okay.

MR. HSIA: Figure 8-2 provides a conceptual three-region model that has been developed from an analytical a fair amount of consideration as --

MEMBER WALLIS: The conceptual isn't much help when you're actually making a calculation.

MR. HSIA: Yes, I understand. Let me finish the sentence, then I'll see if I can understand what this is trying to say. As identified, region one of new reg and two new reg reports, the destructive results example volume instruction of insulation and other debris generated, the size of debris off the break jet force will be considerably different for different types of insulation. Again, Figure A-2 --

MEMBER WALLIS: We know that. We know that.

MR. HSIA: So, this is saying clearly it's conceptual. All we're trying --

MEMBER WALLIS: It's even more confusing, because then you have to give actual numbers for all of these things and you have to show how the zone of influence varies depending on the jet stream --

MR. HSIA: That is the method we are trying to describe in this Reg Guide, saying if you have different insulation, there are different damage pressures for those insulation materials.

Therefore, you need to consider at different distances. Like you quoted, Doctor Wallis, maybe 6 L over D or 20 L over D, that's exactly right. So you cannot just say for my plant I'm going to assume the zone of influence is 20 or 5.

That is not the correct method we're trying to describe here.

CHAIRMAN BONACA: So you have a number of zones of influences, which are material dependent?

MR. HSIA: Correct.

DR. CHANG: Very much so, for the 20 L over D, damage pressure, that is for a much weaker insulation compared to a 5 L over D, such as the so-called --

MR. HSIA: For example, Barsbaeck has, based on our reading, Barsbaeck has one of the worst kind of insulation. At that time, it was just

fiberglass without a very strong jacket.

On the other hand, the reflective metallic insulation would steal a jacket with bindings on it, it would be very strong. So you really need to look at your location and your insulation before you start to go use the zone of influence, whether it's spherical or conical.

CHAIRMAN BONACA: I must say, as I read it, I did not understand that either.

MEMBER WALLIS: I think we need to move on, but we'll come back to this perhaps -- we may not have time, and we just have to be in the letter. I think that even if you can know the damage pressure, then I think you'll find there are inconsistent values from different kinds of research from different places.

And calculate from the damage pressure itself is not something which I'm at all happy about, from your three-region model. So it just changes the devil.

Instead of having spheres that you don't have the size of, it changes the pressures you don't know the value of. So, it's --

DR. LETELLIER: Damage pressure's clearly have to be based on experimentation.

MEMBER WALLIS: Experimentation?

DR. LETELLIER: Yes. And for the database that exists, we have very definite recommendations.

MEMBER WALLIS: The jet pressures? The pressures that are in the two-phase jet?

DR. LETELLIER: Yes.

MEMBER WALLIS: Are based on experimentation, not --

MR. HSIA: That's pressure that can damage the insulation.

DR. LETELLIER: Our recommendations for damage pressure for specific insulation types are based on the record and the data that exist in the data.

There's been extensive testing, and we'd be happy to review that.

MEMBER WALLIS: You measured the pressure on the target?

MR. HSIA: That's correct. That's the pressure on the target.

MEMBER WALLIS: Because you know the pressure in the containment environment?

MR. HSIA: Yes.

MEMBER WALLIS: That's where I have great difficulty with your three-region two-phase conical

jet model. But let's move on. I don't know if you know where it came from.

But if you look at where it came from, you too would have some doubts, I think. Let's move on.

DR. CHANG: Okay, the end consideration of causes is the performance of the ECCS sump - whether the head loss has caused the sump screen will impede the operation of the pump or not for longtime cooling. So that's the bottom line.

MEMBER WALLIS: Okay, they need to calculate that too, don't they?

DR. CHANG: Oh, yes. As a matter of fact, partially the worst break location has very much to do with the head loss across the sump screen.

MEMBER WALLIS: Okay, so in the guidance document that the base is talking about, we have this new Reg CR6224 correlation --

DR. CHANG: Head loss correlation has --

MEMBER WALLIS: One study, which is said to be within 25 percent of the test data. So it looks like a good correlation. Another study, the conclusion was they needed considerable modification.

So, what are you recommending? It's good or it's bad?

DR. LETELLIER: We're recommending it's

application with appropriate parameters based on data. And where --

MEMBER WALLIS: So the licensee has to go through all the database, do his own research, figure out which of these various models and things are appropriate in his plant?

Unless NEI comes up with a very comprehensive analysis of all this somewhat confusing database.

MR. HSIA: It's a fact this is a very complicated and plant-specific issue. We were trying to do a good job throughout the years, trying to cover the bases.

Therefore, we have different data for different applications. We try to test different jets to see which one will be the best one for us to -- for anyone to use to model.

And what NEI will describe remains to be seen. But if they can come out with one generic method, everybody's just going to go with that page so on and so on and come up with the equation, more power to them.

Now, I wish we could do that, but at this moment we're not able to do that.

MEMBER WALLIS: So expecting them to do

research and analysis, which is above a level that you're now capable of doing?

MR. HSIA: If they can do it, yes I'll pass to them.

MEMBER WALLIS: That is a big load for NEI to bear.

MR. MAYFIELD: Let's back up, because that's not what we're saying.

MEMBER WALLIS: Thank you.

MR. MAYFIELD: Go ahead, Bruce.

DR. LETELLIER: Well, I think that we have established a template for quality and standard for experimentation. We have provided the necessary examples for a limited number of insulation types and head loss conditions.

If they're willing to invest the research resources, they certainly know how to proceed. And that's been the intent of our research program, is to establish a minimum level of concern and provide information that's sufficient for us to evaluate the licensee's responses.

We need to have a minimal database for our own needs. And we've focused on the predominant insulation types and the predominant conditions.

MR. MAYFIELD: And the guidance is

structured in that way - it's not a practical matter.

MEMBER WALLIS: The guidance says nothing about the difficulty of making calculations, in fact they don't do it.

DR. LETELLIER: If I can point out, there is a precedent in the BWR resolution, where the guidance was similarly generic and the utilities provided a quite comprehensive --

MEMBER WALLIS: That took a long time.

DR. LETELLIER: It did take a long time.

MEMBER WALLIS: It took ten years, or something like that.

MR. MARSHALL: Again, Michael Marshall, I was the project manager during the BWRs. The BWRs didn't take 10 years to develop that document. It was done in approximately about 18 months or so.

MEMBER WALLIS: But the whole point of the presentation and the resolution of things took quite a long time.

MR. MARSHALL: Right. But as far as coming up with the solutions, the equations and stuff, and the testing and everything they did, it was done on approximately - if I remember correctly, about 18 months.

And again, that facility was done with the

proper testing as such. And again, we provided a template that they've followed and were able to implement using their plant-specific considerations.

MEMBER WALLIS: Thank you, so that's what we're waiting for from NEI?

MR. HSIA: Yes, sir.

MEMBER WALLIS: Okay. Then we need to move on, I think in the instance of time. I don't want to restrict your presentation in anyway.

DR. CHANG: So, I think I can skip maybe -- I sort of described, generally, how the --

MEMBER WALLIS: And there's always an out - if you can't do the analysis, you've assumed 100 percent and that sort of thing.

MR. MARSHALL: Right.

MEMBER WALLIS: And I understand that for many of the Los Alamos studies, a pretty large percentage of the debris actually ended up on the screen for the big breaks.

DR. CHANG: Let me go to the last -- the second to the last view. Graph 13 is on sump screen head loss. Because the sump design of PWRs is very different from the BWRs, so we tried to look at the failure criteria for the ECCS pumps.

And the research showed that for fully

submerged sump screens, the NPSH available in the plant's licensing basis should be the governing criterion for failure.

But for the partially submerged sumps, as I understand, there are a number of plants with only partially submerged sumps. I should call it partially submerged sump screens.

Then NPSH margin may not be the only failure criterion. You have to look at two possibilities. The failure to have enough NPSH margin, will result in the cavitation of the pump.

But another failure mode is the so-called starvation mode. If you have enough head loss across the sump screen, such that the head loss is greater than half of the submerged screen's height, then in that case you will have enough water going into the pump.

MEMBER WALLIS: I think we agreed with that.

DR. CHANG: Right.

MEMBER WALLIS: If I could anticipate your next slide, the problem the Sub-Committee had was that the new research has shown that combinations of fibers and particles can be very effective and very small amounts of debris can block a screen.

And there's a very unexpected, sort of, pressure drop versus stuff calculation where more fibers actually make less pressure drop if you have particular --

MR. HSIA: That's right.

MEMBER WALLIS: Now, this is sort of a new understanding. And in our discussions with you, it turned out that there were certain chemical reactions that hadn't been considered, which could also produce substances which could have an effect on this pressure topic, which might be considerable.

MR. HSIA: Right.

MEMBER WALLIS: Then this doesn't seem to be in the knowledge base, so no NRC reports, and it's only peripherally sort of hinted at in the guide.

And we felt that the chemical effects you bring out, boric acid onto paints, we're putting a lot of material in the pool to raise the pH, and this produces hydrogen and the hydrogen might float debris and so on.

The chemical effects need consideration, and there's some rumor that NEI may not want to proceed until they get better information on some of this chemistry.

DR. LETELLIER: Tony, do --

MR. HSIA: Yesterday, we had a meeting with NRR and NEI. NRR has made it very clear that they would like to continue on current pays -- for the industry to continue on current pays towards resolution of GSI 191.

They would ask the industry to address the issue of chemical effects. The industry at this time is doing a scooping study. Probably, in a matter of a month or so, they will decide whether or not they want to do any additional tests towards that. So, as far as chemical effects, it's --

MEMBER WALLIS: So, one of the things to do, for instance, to improve the situation is to replace all fibrous insulation with reflective foil, which I understand had some fine foil aluminum - lots of fine stuff which in an accident can get blasted out and dumped down into the sump.

Now, I don't know what the reactions are of fine foil aluminum and a large surface area in this kind of environment with very significantly high pH.

MR. HSIA: They certainly, in effect, they would have to consider. They're also stainless steel varieties.

MEMBER WALLIS: Are they going to do the research to find out what happened?

DR. CHANG: As you know, Doctor Wallis, we had a very limited scope on the chemical effect done by LANL and the preliminary tests are completed and we're in the midst of having that report being reviewed by a panel.

As a matter of fact, next Monday we are going to have that review meeting. And we are interested to hear what kind of comments we are going to get from them.

And once we receive that comment, then we will decide what the next step should do.

MR. MAYFIELD: This is Mike Mayfield. Doctor Wallis, you raise an interesting dilemma that we face regularly in research. And that's what's the limit of our responsibility versus responsibility for the industry.

In fact, we get this question regularly from our senior management, from the Commission, and frankly we've gotten it from the Committee over time. I think that Doctor Powers and I have exchanged discussions on this matter.

This is an area where we believe that we have done enough research to show that is, in effect, and while we have not done enough research to say this is how you should -- or one recommended way to deal

with it, we believe that the sum of the feedback we've heard from the utility management is we'd really only like to fix the screens once.

We believe the evidence for this, in effect -- and frankly, it was in effect that Dr. Rosen and Dr. Powers flagged to us sometime back. We believe there's enough evidence to show this is a real effect.

Now, how significant is it --

MEMBER WALLIS: The chemical effect is real?

MR. MAYFIELD: The chemical effect is real. Now, how significant is it depends on very plant-specific details. And that's beyond the level of information we have available to us to sort out on a plant-specific basis.

We felt it was important to flag it in this regulatory guide. And your observation of, well are we putting the onus on the licensees to do the research to develop it?

In part, the answer to that is yes. We have had some discussion, I'm sure we will continue to have some discussions with NRR about how much more do they need to see, in terms of data, to support their evaluations.

MEMBER WALLIS: The concern that I have is

that you'll put out the Reg Guide, which I think is the right thing to do, get things moving, put out this Reg Guide and say, thou shalt evaluate all of these things.

My concern is there are so many things which there isn't much of a technical basis for..

MR. MAYFIELD: Yes, sir.

MEMBER WALLIS: That these folks may come back with some half-baked --

MR. MAYFIELD: Yes, sir.

MEMBER WALLIS: -- analysis, which gets accepted.

MR. MAYFIELD: Well, that's why I --

MEMBER WALLIS: Because nobody knows. And then further research now in progress reveals that it shouldn't have been accepted.

MR. MAYFIELD: Well, that's why -- that is one of the downsides of confirmatory research where I live. The other thing I had said was that we have had, and continue to have, some discussions with NRR about how much more do they need to be comfortable to assess what the licensees are going to bring in the door.

The reason for pushing it forward at this time, to include that loosely worded caveat or flag, is frankly let's put everything on the table at this

time to what level of information we have.

And so we felt like the itch is real, and we needed to flag it in this to the level of detail we can support today, which is to say this is something that should be evaluated.

We will continue to work with NRR, looking at how much more information they need to support an evaluation. But today, we felt like we needed to at least flag the issue in the guide.

MEMBER WALLIS: I think that actually the chemistry is very slightly touched on in the guide, so it parenthetically is that you have to consider environmental and chemical factors.

It doesn't point out that --

MR. MAYFIELD: No, we did put --

DR. CHANG: The debris generated by chemical effects, they are very much like that.

MEMBER WALLIS: It is touched on, but in that sort of parenthetic sort of way, instead of saying this is something important and here are some of the considerations.

And there's nothing about gas evolution and the buoyancy and so on.

MR. MAYFIELD: The level of detail that we put in this is admittedly sparce.

MEMBER WALLIS: So would it be reasonable for us to write a letter that says, yes this thing should go out?

MR. MAYFIELD: Yes.

MEMBER WALLIS: If it gets things moving. And it lays out, although without enough detail on the chemistry, lots of things that need to be considered.

That we have this concern about the knowledge base. Would that be a reasonable thing to say?

CHAIRMAN BONACA: That we've --

MEMBER WALLIS: It might actually help you, knowing that we support what you know to be absent in the knowledge base might help indicate where efforts should be put.

CHAIRMAN BONACA: That's how I think the issue of chemical, for example, concerns may be --

MEMBER WALLIS: Well, we don't know. I mean, Bruce has done tests where it showed that it might well be a concern. And certainly, there's some sort of gelatinous precipate, it's going to effect the screen.

MR. MAYFIELD: Yes. If it manages to come loose, and if it manages to transport, it would be a problem. Those ifs are important. Now, the challenge,

of course, is to figure out exactly how much potentially reactive material is inside containment, and how much of it would actually be exposed to an aqueous environment.

That's a challenge. That's a very plant-specific kind of evaluation. And we felt like, at this stage, it was incumbent on us to at least flag the issue and then let people that have access to the information, meaning the licensees, take a look at it.

MEMBER WALLIS: Your flag is very small.

MR. MAYFIELD: It is a small flag.

MEMBER WALLIS: So we might actually suggest it be bigger. I'm sorry to have picked on these issues, but I think they are the ones that we should focus on in our letter.

Are there other points you want to make? I don't want to limit your presentation, but I think you were moving along anyway.

DR. CHANG: Yes, the last slide is about future research activities. In the near term, we have some calcium silicate head loss test reports. And this is not covered by the new regs 6224 head loss correlation, so we feel that it's appropriate to have some additional testing on this.

MEMBER WALLIS: So the statement in here

that he 6224 needs significant modification is correct. And the other statement that it fits a lot of the data is not really correct?

DR. CHANG: Yes, 6224, that doesn't have the data for all the insulations. And calcium silicate turns out to be -- from a head loss point of view, it's a concern.

And so we think some additional tests should be needed.

DR. LETELLIER: But we are issuing an advisory document at the end of this fiscal year on the head loss properties of calcium-silicate. At a minimum, we'll provide the data that were observed.

And our best recommendations at this time for treating the head loss.

MEMBER WALLIS: This three-region two-phase conical jet model, with numbers on it Figure 17, comes from -- doesn't come from the Sandia work. It doesn't come from the one you referenced.

The only place that I could find it was in a later new Reg that the agency prepared.

DR. CHANG: I think it's in the resolution of USI A-43 documents, is a new Reg report.

MEMBER WALLIS: Right, and my personal view is that it's a complete misapplication of the

Sandia work. Maybe, if my colleagues give me permission, I might actually make a presentation to them on that.

But I just wanted to warn you -- I don't know if you've looked at its origin and seen if you believe it or not.

DR. LETELLIER: That model has been discredited by the Barsebaeck event.

MEMBER WALLIS: Right, it has been.

DR. LETELLIER: In fact --

MEMBER WALLIS: And by practice it's been. But it's in your documents that you've accepted it.

DR. LETELLIER: Are you referring to the knowledge base? Please interpret --

MEMBER WALLIS: But it's there, as being authoritative.

DR. CHANG: The knowledge base report is trying to document order information and pass --

MEMBER WALLIS: But without the critical evaluation, you know, leaves it up to the utilities or NEI to select what's suitable for their purposes.

DR. LETELLIER: Well, that's a fair criticism, that it is presented as authoritative. But it's also intended to be historical. And members of the community that have followed this safety concern

are aware of the improvement in the models.

The Barsebaeck event, we have looked at. And incidentally, we have compared our spherical zone model against that, and shown that it's adequately conservative.

The Barsebaeck event highlighted the fact that material damage is very insulation-type specific. They had -- in fact, it was mineral wall of an aged variety that's very fragile, and not typically used in the United States.

Based on the research work that was implemented for the BWR study, that three-zone model, at least in specifics, with the numbers associated, was discredited and replaced by a better methodology, based on data where you're actually measuring the damage pressures and relating those.

MEMBER WALLIS: But you still have to calculate those damage pressures from a jet model.

DR. LETELLIER: Correct.

MEMBER WALLIS: This discredited model is a jet model, or pretends to be or claims to be.

DR. LETELLIER: The difficulty - particular difficulty with that model is more the qualitative definition of damage, than the calculation of --

MEMBER WALLIS: We'll have to sort this conversation out.

DR. LETELLIER: There's an evolution in thermo-hydraulic modeling as well. And there are a number of alternative models that can be compared and contrasted.

That's an academic exercise it's been ongoing for many years and continues.

MEMBER WALLIS: I don't think it's academic at all to calculate the pressure you need to put into your formula to calculate whether or not insulation is damaged.

DR. LETELLIER: My point is that there are a number of competing models.

MEMBER WALLIS: Yes.

DR. LETELLIER: And they agree to a better or lesser extent to the data, and that's a challenge for numerical modeling.

MEMBER WALLIS: Okay, thank you.

DR. LETELLIER: That continues.

DR. CHANG: Maybe at this point, I think --

MEMBER ROSEN: Let me ask a question about that slide.

DR. CHANG: Yes.

MEMBER ROSEN: The one that's behind you. It says there's a chemical test report due before 10/03. I assume that's 10/31/03?

DR. CHANG: Right.

MEMBER ROSEN: So, we will have -- will we have, when that report's in hand, the answer as to what chemical species are formed, and how -- and what kind of head losses they create in various materials?

The point of this question is, listening to what Mike said about the utility managers, they say they want to fix this once. Well they'll need to know what the effects of the chemicals are.

And if this is the information they need, I think there's no reason for them to have to do it more than once.

MR. MAYFIELD: I'll let Bruce speak to it, but before I do, I would not want to characterize this report that's coming out as the definitive piece of work on chemical effects.

It is not, it was intended to, frankly, build on the issue that you raised, from the TMI experience, and to go back and to say, okay we have the TMI observation.

What do we do with that? How can we recreate that? Can we demonstrate that this sort of

thing can be developed? And, if it's developed, how serious an issue is it, in terms of screen plugging?

The answer is, yes it can be developed. And if it's developed in a sufficient quantity, that's a problem. So, I wouldn't want to oversell what you're going to find in that October report.

MEMBER ROSEN: So, you're suggesting, perhaps, that there will be more chemical work done after October?

MR. MAYFIELD: I'm suggesting that somebody's going to have to do a lot more chemical work. And the discussion we've had about it, is who's going to do it and how much more is really needed.

MEMBER WALLIS: So when can you decide what the utilities should do?

MR. MAYFIELD: Well, Doctor Wallis, that's -- again, the problem that I face in managing work, confirmatory research, is that I'm constantly running behind when my colleagues at NRR have to make a decision.

MEMBER WALLIS: So, it's not your -- it's the NRR folks, it isn't you.

MR. MAYFIELD: No, sir, well, they're the ones that find themselves having to ultimately take a deep breath and make a decision. And they look to us

to provide them additional information to support that. But that's the nature of where we are.

CHAIRMAN BONACA: I had a question, with regards to this near-term and long-term work. I mean, now if we publish this Reg Guide 1.82, how are you going to document this new information?

Is it going to be purely knowledge, added knowledge?

MR. MAYFIELD: It would be added knowledge. And if we find something that we believe takes -- makes sort of the next major step in either, oh by the way there was an error in this guide, or here's some additional information, we'll revise the guide again.

Obviously, we've been willing to revise it in the past. This is a --

CHAIRMAN BONACA: So, basically, you're planning to have a second document? This is --

MR. MAYFIELD: We would almost certainly publish additional new reg reports to document this as we go along. And, frankly, we can get that information out through the publication of a new reg and then through various generic communications that NRR has.

So the information can be made available fairly quickly. To modify a Reg Guide obviously is a

more time-consuming process.

MR. JOHNSON: Yes, Mike Johnson, just to add... You know, we are anxious, obviously, anxiously awaiting what the report says, what the peer review thinks of the report, what the final report says, as is the industry.

One of the things that he industry raised at the meeting that we had with them, where they committed to continue to pursue resolution of GSI 191, and to also look at this issue once it becomes more well-defined.

We're all anxious to see what comes out, to make sure that we can approach both of these issues and not delay resolution GSI 191 while we, again, figure out what's going on with the chemical effects precipitation.

And again, hopefully the industry can take only one fix. They would like to, obviously they've told us they'd like only to make one fix. But they also recognize that, as we figure out what we have to do to get our hands around this issue, they might actually have to do more than one fix.

MEMBER WALLIS: With regards to the chemistry, we saw some preliminary results of chemical work, which were very interesting. And the comment of

the Sub-Committee was these were very interesting, but they don't really duplicate the chemistry in the plant.

Yes, there's zinc in the paint, but it's not elemental zinc, it's probably zinc chromate or something - it's a zinc in some form other than disks of zinc.

And if you do an experiment with disks of zinc, you're not really duplicating what happens to paint, that the temperatures, the pH, the chemical constituents and so on, should be realistic, as far as the plant goes.

And the constituent, you're likely to find there. And that sounds like a fairly extensive program.

MR. MAYFIELD: I agree. To really pin this down and develop all of the data that you would like to have, is a significant undertaking.

MEMBER WALLIS: Thank you. Yes.

DR. CHANG: In the long-term, we're talking about up to September of next year, we are going to do some additional test, such as latent debris collected from volunteer plants, such as dirt, dust, rust, all those things you can gather from operating debris.

MEMBER WALLIS: And that's going to be put into the chemical test too?

DR. LETELLIER: The primary objective is to characterize the hydraulic properties of this debris, as a particular. In the BWRs, we had iron oxide as a predominant particular source.

And we would like to characterize the P's in a similar way.

DR. CHANG: And we are going to do a head loss test on those debris.

DR. LETELLIER: The hope of the characterization is to come up with a recipe for screening, sieving, mixing up additional quantities that are useful for head loss testing.

The reason this research was started in the beginning is one of our early attempts at creating dust was to screen -- sweep up the concrete lab at the University of New Mexico and dump that into the bed.

And people criticized - the industry, in particular, was not pleased with that, so... We're going back to look at the composition of actual resident material.

DR. CHANG: And it's possible that we're going to do some HPSI frontal valve plugging tests. And in the February/March timeframe next year, there

will be an international workshop, in Albuquerque, New Mexico, on the PWR clogging issue, right?

DR. LETELLIER: Correct.

MEMBER WALLIS: Are you going to do any internal clogging tests? I mean, none of this debris -- there's a pretty course screen and a big pump and a big HPSI valve and all.

It gets into the radi-coolant system, some particles. And the clogging of the spaces and the fuel and the flakes, and so on...

DR. LETELLIER: I think the high pressure safety injection, the throttle valve has been identified as one of the smallest internal gap tolerances, that's why we're --

MEMBER WALLIS: But the fluid's whipping through there, isn't it? It's going to carry -- there are pure fluids whipping through there?

DR. LETELLIER: It is.

MEMBER WALLIS: Right, so... it's not just a question of size, it's a hydraulic conditions.

MEMBER ROSEN: But I don't think you answered Doctor Wallis' question about the fuel.

MR. MAYFIELD: I was just going to jump on that. One of the -- this international workshop, I'm probably at the bottom of. I met with the Germans

about a year ago to talk about a range of issues and the sump blockage issue was one of them.

They discussed in exactly this issue, and they've concluded that that's something that they are concerned about for their configurations. The potential for debris to pass through the system and lodge in various places, as you go through the core.

And that's an issue that they have been actively pursuing. And our intent is to build on the work that they have been doing. But we also know that there has been other bits of work done by very competent laboratories around the world, and we wanted to capitalize on that work, rather than re-invent the wheel every time.

So, we have had, and continue to have, a dialogue with those organizations to build on their knowledge and understanding. And this international workshop is one that we pushed for, to try to get all of the people, or at least the major players together, at one time to discuss in detail the work they're doing and they're finding.

And then we'll roll that information into the next steps that we're taking. We had frankly -- I'd been pushing T.Y.'s predecessor, who had mysteriously shows up down here with the staff now -

I'd been pushing him to have this workshop significantly earlier.

And just the logistics, it wasn't a practical matter. So, we have this thing scheduled now. We know there's a lot of interest in pursuing it. And for our application, we'll see how significant the fuel issue really is.

It is something we are aware of, and we're looking to capitalize on that international data to pursue it.

MR. ARCHITZEL: This is Ralph Architzel, from NRR, if I can just interject for a second. Separate from GSI 191, downstream blockage issues have been raised in the bulletin, and are planned to be raised on generic letter, so that it's not a part of GSI 191 per se, but it is part of the documentation going with the bulletin.

Those licensees -- that one licensee that gave us category one response did address the fuel blockage inside the vessel. That's one of the examples listed.

The other plants will be asked to address that. It's not part of the NEI guidance document, it's considered an engineering issue that should be addressed by licensees with a resolution of the future

generic letter, not GSI 191.

But I wanted to point that out that that's an issue.

MEMBER ROSEN: It's not in the NEI document because there are so many different fuel types?

MR. ARCHITZEL: It's not in the NEI document because NEI had a scope. And their scope was to address GSI 191 and they chose not to address downstream blockage, upstream blockage, structural integrity of the screens.

Things like that are considered engineering issues.

MEMBER ROSEN: How could they -- if their scope was GSI 191, why isn't this part of it?

MR. ARCHITZEL: This isn't part of GSI 191, GSI 191 was not blockage inside the fuel channels and things like that. I'm saying that's not what GSI -- some performances what GSI 191 was.

MR. MAYFIELD: One of the issues that we struggle with in managing the generic safety issue program is what we call scope creep. And the issues simply never go away, because there's always the next piece.

So we've chosen to go at this in a

somewhat different way. And one of the discussions I've had with Mr. Thadani, goes to why aren't we opening yet another generic safety issue?

And that's an open discussion that we'll take on.

MEMBER ROSEN: That's perfectly acceptable. It was just a question of definition. I mean, the physical world doesn't know that these effects have separated.

MR. MAYFIELD: That's exactly correct. This is a bureaucratic issue.

DR. CHANG: At this point, may I suggest that let Bruce present his slice on the ZOI. Hopefully that will answer some of your questions.

MR. MAYFIELD: Let me ask this somewhat differently. Does the Committee wish to pursue the technical details on the zone of influence?

MEMBER WALLIS: I don't think this is the place to do it.

DR. LETELLIER: We would be happy to meet with you privately, or teleconference.

MR. MAYFIELD: Or we can do it through another Sub-Committee meeting - however the Committee would choose to go at that. I go the distinct impression from the earlier discussion that there are

some substantive technical questions at a fairly low level of detail, or high level, however you want to look at that.

MEMBER WALLIS: Yes, but we have to write the letter, rather than engage in consulting with you guys. So, I think we're going to have to put some of these technical questions in the letter.

MR. MAYFIELD: That's obviously a fair approach. We do continue to believe it's important to get this guide on the street. I understand your concern.

MEMBER WALLIS: That's the key issue, I think. Get it out there, in spite of the fact that it's tremendous amount of work needed to be done to really meet the requirements of it.

MR. MAYFIELD: Right, and we continue to believe that's important and we would hope to get a letter from the Committee that would support moving forward.

MEMBER POWERS: Let me ask, Mike, just a question a little bit about the chemistry issues that have come up in regards to what's in the sump and what can produce and things like that.

You kind of have a Duke's mixture of junk, potentially present here. You've got some plans to try

to limit that somewhat below 92 possible elements, I take it.

MR. MAYFIELD: That'd be nice.

MEMBER POWERS: Yes, have you taken something like YQ or some of their aqueous equilibrium code and said, okay I don't know that I have equilibrium but what do I have if I put this junk into a hot sodium hydroxide solution, maybe with sodium phosphate in it, or potassium phosphate in it in some cases.

MR. MAYFIELD: The answer to that is, no we have not pursued that. The one issue, and the Committee had raised this, that the observation from TMI, which obviously is something we hadn't picked up.

We went back, did enough testing to convince ourselves no we can't quite make it go away. And then the next question is, well how much more do we need to do, in responding to Doctor Wallis.

It's a big undertaking to really get your arms all of the way around it. The approach you're proposing is one of the things, whether it's that particular code or another approach, that's one of the things that you would have to pursue, it seems to me.

But it's -- the exact structure of the research program that you'd put together to take that

on, is plainly something we haven't worked all the way through.

MEMBER POWERS: Sure. One of the things that I would tend to push back on, is when somebody tells me, oh the chemicals that you put into this are not exactly precisely the same particle size, method of manufacture or chemical form, of the chemicals that I think I have in plants.

For instance, I think particularly the zinc that may come from a paint that by the time you take your zinc disk and put it into sodium hydroxide solution, it's pretty warm.

The zirconium oxide, hydroxide that you get off that, pretty well can't tell where it came from. And --

MEMBER WALLIS: Zinc hydroxide, right?

MEMBER POWERS: Zinc oxy-hydroxide. It's an interesting material because it's transient in nature. And it even gets modified further if pour boric in there, it's more gelatinous material.

MEMBER ROSEN: I guarantee you that the boric acid erodes.

MEMBER POWERS: And, I mean, those kinds of things would make your chore, characterizing the chemistry, impossible, okay? So you need -- whether

you do the experimental work yourself, or you are in the position of evaluating the product or the licensee's work on the chemistry, you need some sort of a computational vehicle to say, is this in the realm of reasonableness, from a chemical point of view?

Or, is this something very strange and weird? It might be worthwhile to look into that.

MEMBER KRESS: You have to be a little careful to interpret the equilibrium quotes at like -- if you can get a kinetics code, it'd be a lot better.

MEMBER POWERS: Tom, quite frankly, in the history of looking at these things, what I know is it's really easy to get heterogeneous things that are weird, in reality, that you don't get equilibrium on solution kinetics, and these things are pretty fast. But the precipitates can be weird on you.

MEMBER KRESS: That's the sort of thing I was worried about. You'd get an intermediate reaction that precipitates, and you won't know that with an equilibrium code.

MEMBER POWERS: I mean the world, in this computational modeling, has undergone some substantial evolution, largely because of places like WIPP and Yucca Mountain, because they have the same problem.

They have to predict what's in these rock pores, precipitates out and blocks them and absorbs things and stuff like that. And at least it gives you a shot at understanding.

MEMBER KRESS: I agree, it'd be a good way to start, the easiest way to start.

MEMBER POWERS: It's the cheapest and easiest way to start, especially if you're starting off well I've go 92 elements.

MR. MAYFIELD: We would certainly be willing to talk with the Committee about the approach that we would take a look at. Again, this has been an open dialogue with NRR about how much further they would like to see us go, to be able to support them and their reading.

MEMBER POWERS: I guess I have two points here. One of which is, I don't think you're going to be able to wash your hands completely of the chemistry problem, just because you're going to have to review what somebody does.

MR. MAYFIELD: I don't think we can walk away from it. The question is, how clean can I get my hands?

MEMBER POWERS: I guess I would side with you. I'd keep myself as far out of the laboratory as

I could.

MR. MAYFIELD: They don't keep me very close anymore.

(Laughter.)

MEMBER POWERS: Why? I understand that, but just because I suspect you will find that plants differ in the junk that's on the floor.

MR. MAYFIELD: Yes.

MEMBER WALLIS: I guess the Sub-Committee felt the opposite way, that you had to be in the lab, you had to do some tests with some real paint and some real temperatures and pH's and things, and get some idea of what these things might do.

MEMBER POWERS: I mean, quite frankly, that research on paint, the NRC has been intimately involved in pretty extensive. I mean, we know a lot about how paint behaves, because in these accident environments, simply because it also tends to be a pretty good absorber of iodine.

And I think there's a lot you can get, without actually going and putting salts in solutions.

MR. MAYFIELD: I would also suggest that it's not just paint. There's all manner of conduits and cable trays and other bits and pieces that could be of concern.

MEMBER POWERS: And you've got some real amazing things when you throw a little boric acid into a little concrete dust. Because then you get this calcium borate - I think it's called whistlelight, or something like that, that's just amazing stuff.

MEMBER WALLIS: Why is it amazing?

MEMBER POWERS: Oh, it's long strings.

MEMBER WALLIS: So it clogs, then? The long strings would tend to clog things.

MEMBER POWERS: It makes -- it's weird stuff.

DR. LETELLIER: In fact, we did add calcium to our basic stock solution, to account for concrete ablation.

MEMBER POWERS: You should have gotten a little bit of nice gelatinous precipitate out of it.

DR. LETELLIER: Indeed, we did.

MEMBER POWERS: Yes, you got whistlelight.

DR. LETELLIER: I'd like to correct a couple of misperceptions of Doctor Wallis. In fact, we did test zinc paint chips, which is a representative material.

I think the biggest deficiency of our quiescent immersion test is the fact that it's not a turbulent flowing solution. I think we may be seeing

some surface crystallization that might not occur.

MEMBER WALLIS: This was, I think, my colleague who isn't here, Doctor Ford said that the zinc that you tested wasn't quite the same as the chromate primers and things that you find in the real plants.

DR. LETELLIER: That is a fact that we're testing --

MEMBER WALLIS: All right, so it wasn't the same.

DR. LETELLIER: But we're testing metallic zinc granules.

MEMBER WALLIS: Right, it's not the same thing.

DR. LETELLIER: That's correct. We did our best effort at reproducing the pH conditions. The temperature is a little bit low, thinking that if we can induce this, or establish this as a concern at low temperature, then certainly it is a concern at higher temperature.

MEMBER POWERS: Warm that solution up in zinc chromate, it turns into oxy carbonate in a thrice plus a little chromus oxide.

MEMBER WALLIS: Can we wrap this thing up? I'd be very happy to meet with you folks in the office

here. Anybody else?

MR. MAYFIELD: Let me try to close it out, then, Doctor Wallis. Again, we appreciate the opportunity to come before the Committee again this afternoon.

We would welcome your insights, both individually and whether it's through the Sub-Committee or the full Committee, we would very much appreciate a letter that would endorse moving forward on this.

And we would be interested in the list of issues that you believe we need to work more on. And with that, unless you have further questions, that concludes our presentation.

MEMBER WALLIS: Does anyone on the Committee want to speak up? Then I hand it back to you, Mr. Chairman.

CHAIRMAN BONACA: Okay, well thank you. I thank you very much for the presentation. And I think what we're going to do now is take a break - some of us have been at it since 2:30 p.m.

And then I think we will have the presentation from Nourbakhsh should be tomorrow, because we really don't have time today. What I would like to do is go down the table and discuss at least

two letters for which I think we need to provide the writers with inputs from the Committee. One is the one on -

MEMBER POWERS: The alpha and the omega.

CHAIRMAN BONACA: They may be.

(Laughter.)

CHAIRMAN BONACA: One is the one on heavy loads. I think one is on the PRA. Okay, so you already knew what we have in mind? Okay, all right, and is there any other letter for which you believe we need to provide some input?

MEMBER SIEBER: They're printing the one on 186.

CHAIRMAN BONACA: Yours?

MEMBER SIEBER: Yes.

CHAIRMAN BONACA: Okay, what about the one on--

MEMBER KRESS: I already got --

CHAIRMAN BONACA: You already got feedback yesterday, I thought. So I was worrying about mostly the one from Jack, the one from George and the one from Vic. We'll be back in here in 15 minutes, 10 after 6:00 p.m. Thank you.

(Whereupon, the foregoing matter went off the record at 5:48 p.m.)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

September 30, 2003

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DRAFT FINAL REVISION 3 TO REGULATORY GUIDE 1.82, "WATER SOURCES FOR LONG-TERM RECIRCULATION COOLING FOLLOWING A LOSS-OF-COOLANT ACCIDENT"

Dear Chairman Diaz:

During the 505th meeting of the Advisory Committee on Reactor Safeguards, September 10-13, 2003, we met with representatives of the NRC staff to discuss the draft final Revision 3 to Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Ref. 1). Our Subcommittee on Thermal-Hydraulic Phenomena also reviewed this matter during its meeting on August 19, 2003. We previously provided a letter, dated February 20, 2003, concerning an earlier draft of this guidance. Regulatory Guide 1.82 (RG 1.82) is being revised to enhance the debris blockage evaluation guidance for pressurized water reactors. We also had the benefit of the documents referenced.

Recommendations

1. Draft final Revision 3 to RG 1.82 should be issued in order to facilitate licensee response and the resolution of technical issues. In addition, the staff should carefully review implementing guidance being developed by the Nuclear Energy Institute (NEI) because of the issues identified, the complex phenomena involved, and the need for more accurate plant-specific assessments.

2. The knowledge base report (Ref. 2) is a compendium of research results relevant to the problem, but it is confusing and it cannot be used directly as guidance for the analysis of sump blockage. Acceptable methods should be developed for use in satisfying the functional requirements described in RG 1.82.

3. An adequate technical basis should be developed to resolve the issues related to chemical reactions.

4. The staff should consider the possibility that the uncertainties associated with the calculational methodology may be so large, or that strainers may prove to be so susceptible to debris blockage, that alternative solutions may be required to ensure long-term cooling. This might involve, for example, changing the types of insulation used within containment or implementing diverse means of providing long-term cooling.

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5. The staff should investigate a risk-informed approach to sump screen blockage.

Conclusions

- The technical basis for analyzing the phenomena described in RG 1.82 is not mature, the available information is inconsistent, and the knowledge base is evolving. Therefore, it is likely that the licensees' responses will be disparate and difficult to evaluate unless more consistent guidance is developed.

- The zone of influence (ZOI) models need revision and resolution of inconsistencies.

- Neither RG 1.82 nor the knowledge base report (Ref. 2) gives adequate consideration to chemical reactions.

Discussion

The sump screen blockage issue has a long history, dating back to the 1979 unresolved safety issue (USI) A-43. More stringent requirements have been developed as incidents or new knowledge revealed a need. These are reflected in various Bulletins, Generic Letters, and earlier revisions to RG 1.82. The case of boiling water reactors (BWRs) was revisited after the resolution of USI A-43 in 1985 because of several events, such as the one at the Swedish Barsebäck Nuclear Plant, Unit 2, in 1992, which demonstrated that larger quantities of fibrous debris could reach the strainers than had been predicted by models and analysis methods developed for the resolution of USI A-43 (Ref. 2). The BWR issue was resolved by installing large-capacity strainers in response to Bulletins 93-02 and 93-03. The strainers were designed on the basis of a BWR Owners Group report, NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," November 1996, which was approved by the staff.

The results of recent parametric study (Ref. 3) of 69 pressurized water reactors (PWRs) revealed that following a large-break LOCA, sump screen blockage was very likely in 53 of them. The same report stated that preliminary findings suggest that two-phase jets with a stagnation pressure of 1400 psia can inflict significant damage at distances much farther away than those measured in either USI A-43 studies or BWR air-jet impact tests program. Recent research has led to the discovery that very thin beds of fibrous insulation of the order of 1/8 inch thickness, in combination with particulates, can effectively block a sump screen. A risk study that supported the parametric study suggested an increase in the total core damage frequency (CDF) of an order of magnitude or more (Ref. 2). These studies were qualified with the caveat that many features of the problem are plant specific and, therefore, must be evaluated at that level. There appears to be sufficient evidence that new NRC guidance is necessary and appropriate action by PWR licensees may be needed.

Revision 3 to RG 1.82 describes the functional performance requirements for water sources that support long-term cooling. It also describes the main phenomena that are to be considered in the analysis of the performance of these sources, although it makes only general reference to chemical phenomena that may be important. Revision 3 to RG 1.82 should be issued in order to facilitate licensee response and the resolution of technical issues.

NEI is developing an implementing guidance document for licensees. Because of the many phenomena involved, and the significant plant-dependent nature of their manifestation, the staff will have to carefully review the NEI guidance and may need to perform confirmatory research.

While the revised RG 1.82 provides an extensive description of the phenomena of interest, it has little to say about the methods to be used for analyzing such phenomena. The major source of information on possible approaches has been the knowledge base report (Ref. 2) prepared recently by the Los Alamos National Laboratory. While this report comprises a compendium of research results obtained over several decades, these results are sometimes inconsistent and some have been superseded by recent work. The report does not clearly identify which results are valid, does not resolve apparent inconsistencies in the various studies, does not present a synthesis of validated methodologies that can be applied to actual plants, and provides little perspective to guide the user in the choice of appropriate quantitative methods.

For example, the production of debris is considered to occur in a ZOI. This is a useful concept, but for practical purposes, quantitative methods for describing the ZOI are necessary and Reference 2 provides several conflicting approaches. On page 3-25 it states that in a conical jet the centerline stagnation pressure is essentially constant at a distance of about 5-7 pipe diameters, at approximately 2 ± 1 bars. Figure 3-17 shows stagnation pressures between 3.5 and 5.5 bars in the same region. Both of these results originate from methods developed to resolve USI A-43, which were found to underestimate the Barsebäck damage. Results of recent studies show a pressure of about 11 bars in this same region. Page 3-6 states that the ZOI associated with prototypic two-phase (steam-water) jets is larger than the ZOI indicated by air jet simulated tests. Combining this with the statement on page 38 of NUREG/CR-6762, Volume I that single-phase air jets inflict significant damage to fibrous insulation types at a distance of 60 pipe diameters, one would conclude that the zone of influence is much greater than indicated in Figure 3.17. If licensees were to use such disparate information, we would anticipate the same variability in application of methods that was apparent in the BWR submittals.

During our meetings, the staff stated that the ZOI could comprise a large fraction of the entire containment. This does not seem consistent with the rather small ZOI shown in Figure 3-18 of the knowledge base report (Ref. 2). This figure is based on a set of spheres with the same volume as the zones shown in Figure 3-17, which is claimed to be a conical jet model originating from the work (Ref. 4) on jet loads reported by the Sandia National Laboratories (SNL) in NUREG/CR-2913, Rev. 4. The figure does not appear in the SNL report, but is actually Figure 3.25 of NUREG-0897, "Containment Emergency Sump Performance: Technical Findings Related to Unresolved Safety Issue A-43," 1985 (Ref.5). Use of this figure for estimating loads on containment structures appears to be a result of a misapplication of the SNL work, which considered the impact of a two-phase jet, issuing from a round break of diameter (D), on a large flat target perpendicular to the axis of the jet and a distance (L) away. The pressure distribution was computed on the target, as a function of radial distance, (R), from the axis. The stagnation pressure on the axis was lower than the original stagnation pressure of the jet because a shock wave occurred before impact on the target. This shock wave was the only mechanism of energy dissipation. Figure 3.17 in the technical basis report (Ref. 2) was constructed from contours of constant pressure on the target as (L) was varied.

This approach to computing a ZOI has two major errors. The first is the use of pressure distribution on a flat target to characterize the pressure felt by an object (such as a pipe) inserted into the same flow field when the target is there. The pressure falls away from the stagnation point on the target because of the large velocity of the fluid along the plate. However, if a pipe were placed on or near the plate at some radius, the fluid coming to rest at the stagnation point on this pipe would achieve a high pressure, comparable with the stagnation pressure at the axis of the target, as it was brought to rest. Moreover, the fluid that is diverted by the plate and disperses to the sides over a cylindrically-shaped area still has a very high velocity. For example, Figures 4.10 to 4.14 of the SNL report (Ref. 4) show that, in this example with L/D =2, at a radius of 5 diameters, the fluid flowing along the plate has a speed of about 2500 ft/sec while the fluid flowing along the plate from which the jet issued has a speed of about 3500 ft/sec. This latter fluid has not suffered a shock and has lost none of its energy. The result is a disc-shaped jet with an area that is 80 times the area of the original jet issuing radially into the surrounding space. Should the part of the jet that has not passed through a shock strike an object, the pressure load, according to the SNL model, would only be mitigated by whatever shock wave occurred in front of that object. Should the jet be focused by passing between suitable structures, it could conceivably recover most of its original stagnation pressure of 150 bars. The point is that even if there is a flat target in front of the jet the loads on other structures are not determined solely by the pressure distribution on that target.

The second misuse of the SNL work is to interpret the contours of static pressure on the target plate as being representative of the stagnation pressure distribution in a jet when the plate is not there. The reduction in radial static pressure over the plate is determined by the radial velocity which is not the same as in a jet in the absence of a target. Moreover, the stagnation pressure distribution in the jet is what is needed to determine the maximum pressure on structures, not the static pressure, and it is uniform until the flow passes through a shock wave. In fact, with the assumptions of the original SNL model, the stagnation pressure is uniform everywhere, to any distance, until a shock wave is passed through by the fluid. To assess the pressure exerted on an object, one would have to compute the flow field for a free jet and evaluate the strength of the shock wave ahead of that object when placed in this field. In practice, in a real containment, there will be shock reflections from multiple objects, redirection of the flow, and possible refocusing of the energy.

Given these concerns, the NRC staff should reevaluate the basis for establishing a ZOI. That basis should be quantitatively related to actual damage observed in plants and in experiments designed to assess the actual damage observed in various flow fields. These events and experiments have been reported (Ref. 2) but have not been used to develop validated practical prediction methods.

Another concern is the lack of consideration given to chemical effects, in both RG 1.82 and the knowledge base report. A hot, acidic, borated, two-phase jet has the potential to react chemically with paints, coatings, insulation, and other materials, particularly those incorporating aluminum and zinc. When the hot, borated water drains to the pool, it is dosed with alkaline material to create a high pH in the pool. In the presence of zinc, this is known to lead to the production of zinc hydroxide with concomitant evolution of hydrogen. Results of some preliminary experiments performed by LANL indicate that several other precipitates may be formed, some of which have a gel-like or sticky consistency that could exacerbate the potential for screen blockage.

In addition, hydrogen evolution in the pool is likely to affect the settling of materials that are heavier than water. A zinc particle, for example, will sink in pure water; however, if a reaction produces hydrogen bubbles that stick to the surface of the zinc particle, the particle may become buoyant and rise to the surface, probably eventually sinking again as the bubbles are released, with the cycle repeating. Similarly, a sediment of fibrous debris could be rendered buoyant by gas bubbles released within it.

The chemical kinetics of the reactions of concern may be too slow to influence sump blockage. However, this needs to be shown by definitive analysis and testing. Moreover to the extent possible, such testing should be performed under the conditions expected in an actual plant.

RG 1.82 gives passing reference to chemistry in Sections 1.3.2.6 and 2.3.1.8, which state that debris created by the resulting containment environment (thermal and chemical) should be considered in the analysis. However, in response to a public comment, the staff acknowledged that there are no NRC-published references pertinent to consideration of these chemical reactions. While RG 1.82 discusses effects of buoyancy on debris transport, it does not mention buoyancy induced by the release of gas by chemical reactions.

The knowledge base report describes many experiments, most of which were conducted under laboratory conditions, designed to investigate the transport of debris. These are useful sources of information; however, the report presents many qualifications of these results, particularly in view of the variety of phenomena involved in an actual plant. For example, one area of concern is the potential for debris to block flow paths to the sump before reaching the pool; these paths are numerous and vary significantly from plant to plant.

Knowledge about the head loss to be expected on sump screens is evolving, with recognition that the combination of fibrous and particulate materials can produce unusual effects. Again, this knowledge base needs to be consolidated into a form that is less susceptible to misinterpretation by readers. For instance, page 7-6 of the knowledge based report (Ref. 2) states that the NUREG/CR-6224 correlation will need considerable modification, whereas page 7-29 appears to endorse the same correlation with the statement that its predictions were within ± 25% of the test data.

There is also a need to synthesize this information into practical methods of prediction. The forthcoming NEI guidance should help in this regard.

As we discussed in our letter dated February 20, 2003, there is a possibility that the assessment of the blockage of the sump strainer may be subject to such large uncertainties as to be intractable, and alternative solutions may be required to ensure long-term cooling. These might involve, for example, using active sump screen systems, changing the types of insulation used within containment, or implementing diverse means of providing long-term cooling, including using additional water sources to extend the injection phase. Section 1.1.4 of Revision 3 to RG 1.82 discusses the use of active sump screen systems, but these may be only one of several possible alternatives that should be considered to ensure long-term cooling.

PWR sump blockage is an issue for which the design-basis accident approach may lead to unnecessary conservatism. A risk-informed approach may be appropriate in which the design-basis requirement to maintain effective long-term recirculation cooling would be retained, but risk information would be used to establish an acceptable approach to comply with the requirements.

The quantification of the sump blockage issue is an excellent example of where risk information can be applied to design-basis accident issues to the benefit of the public and the licensees. The staff should explore the feasibility of a risk-informed approach to sump screen blockage.

Sincerely,

/RA/

Mario V. Bonaca
Chairman

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May 19, 1999

The Honorable Shirley Ann Jackson
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: THE ROLE OF DEFENSE IN DEPTH IN A RISK-INFORMED REGULATORY SYSTEM

During the 462nd and 461st meetings of the Advisory Committee on Reactor Safeguards, May 5-8 and April 7-10 1999, we discussed issues identified in the Staff Requirements Memorandum dated March 5, 1999, concerning the appropriate relationship and balance between probabilistic risk assessment (PRA) and defense in depth in the context of risk-informed regulation. We previously discussed this matter with the Commission during our meeting on February 3, 1999.

We are attempting to identify pitfalls that may exist along the path the Commission is taking toward risk-informed regulation so they may be addressed in a timely manner. We have communicated previously on the need for plant-specific safety goals that are practical for licensees to evaluate, the need for risk assessments for all modes of plant operation, and the need for research to support further use of risk information in regulatory activities. Several ACRS members, working with an ACRS Senior Fellow, have produced the attached paper in which two views of defense in depth are discussed along with a preliminary proposal regarding its role. Here, we further discuss the role that defense in depth should have in a risk-informed regulatory scheme.

Our motivation for this report has arisen because of instances in which seemingly arbitrary appeals to defense in depth have been used to avoid making changes in regulations or regulatory practices that seemed appropriate in the light of results of quantitative risk analyses. Certainly, we have seen defense in depth used as a basis for delaying changes in the existing regulatory practices:

- there has been reluctance to develop new, risk-informed limits on leakage from steam generator tubes because these are part of the defense-in-depth barriers,
- the development of extensions of the Regulatory Guide 1.174 process to define criteria for risk-informed revisions to 10 CFR 50.59 has been delayed because of defense in depth issues,
- the development of graded quality assurance measures has been overly conservative because of concerns about the imputed importance of quality assurance to defense in depth, and
- the development of regulatory requirements on software-based digital instrumentation and control systems was delayed because of concerns related to defense in depth.

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We are concerned that arbitrary appeals to defense in depth could inhibit the effective use of risk information in the regulatory process. At the same time, we are mindful that risk analyses are not perfect. Defense in depth can be an effective means for compensating for any weaknesses in our ability to understand the risks posed by nuclear power plants.

As discussed in the attached paper, the defense-in-depth approach to safety arose in an earlier time when there was less capability to analyze a nuclear power plant as an integrated system. Subsystems were designed such that the necessity and sufficiency of defense in depth could be determined from experience and through exercising engineering judgment. Defense in depth was a design and operational philosophy that called for multiple layers of protection to prevent and mitigate accidents. Its practical implementation was most often associated with control of initiating event frequencies, redundancy and diversity in key safety functions, multiple physical barriers to fission-product release, and emergency response measures. This philosophy has been invoked primarily to compensate for uncertainty in our knowledge of the progression of accidents at nuclear power plants.

Improved capability to analyze nuclear power plants as integrated systems is leading us to reconsider the role of defense in depth. Defense in depth can still provide needed safety assurance in areas not treated or poorly treated by modern analyses or when results of the analyses are quite uncertain. To avoid conflict between the useful elements of defense in depth and the benefits that can be derived from quantitative risk assessment methods, constraints of necessity and sufficiency must be imposed on the application of defense in depth and these must somehow be related to the uncertainties associated with our ability to assess the risk.

We believe that two different perceptions of defense in depth are prominent. In one view (the "structuralist" view as described in the attached paper), defense in depth is considered to be the application of multiple and redundant measures to identify, prevent, or mitigate accidents to such a degree that the design meets the safety objectives. This is the general view taken by the plant designers. The other view (the "rationalist"), sees the proper role of defense in depth in a risk- informed regulatory scheme as compensation for inadequacies, incompleteness, and omissions of risk analyses. We choose here to refer to the inadequacies, incompleteness, and omissions collectively as uncertainties. Defense-in-depth measures are those that are applied to the design or operation of a plant in order to reduce the uncertainties in the determination of the overall regulatory objectives to acceptable levels. Ideally then, there would be an inverse correlation between the uncertainty in the results of risk assessments and the extent to which defense in depth is applied. For those uncertainties that can be directly evaluated, this inverse correlation between defense in depth and the uncertainty should be manifest in a sophisticated PRA uncertainty analysis.

When defense in depth is applied, a justification is needed that is as quantitative as possible of both the necessity and sufficiency of the defense-in-depth measures. Unless defense-in-depth measures are justified in terms of necessity and sufficiency, the full benefits of risk-informed regulation cannot be realized.

The use of quantitative risk-assessment methods and the proper imposition of defense-in-depth measures would be facilitated considerably by the availability of risk-acceptance criteria applicable at a greater level of detail than those we now have. Development of the additional risk-acceptance criteria would have to take into consideration safety objectives embodied in the existing regulations. For example, risk-acceptance criteria are needed to meet the Commission's safety objectives with respect to worker health and environmental contamination and to meet additional public health and safety objectives [e.g., total fatalities, land interdiction]. All of these may not be currently reflected in conventional risk assessments.

We believe that a key missing ingredient needed to place quantitative limits on defense-in-depth measures is acceptance values on the level of uncertainty for each safety objective. Setting such acceptance values is a policy role, very much like setting safety goal values. The uncertainties that are intended to be compensated for by defense in depth include all uncertainties (epistemic and aleatory). Not all of these are directly assessed in a normal PRA uncertainty analysis. Therefore, when acceptance values are placed on uncertainty, these would have to appropriately incorporate consideration of the additional uncertainties not subject to direct quantification by the PRA. These considerations would have to be determined by judgment and expert opinion. As a practical matter, we suggest that the acceptance values be placed on only those epistemic uncertainties quantifiable by the PRA but that these be set sufficiently low to accommodate the unquantified aleatory uncertainties.

When acceptance values have been chosen as policy for the regulatory objectives and their associated uncertainties, it would be possible to develop objective limits on the amount of defense in depth required for those design and operational elements that are subject to evaluation by PRA. To do this, it is necessary to incorporate the effects of the defense-in-depth measures into the PRA uncertainty analysis and the designer or regulator must be able to adjust the defense in depth until the acceptance levels for the regulatory objectives and the acceptance values for the associated uncertainties have both been achieved.

The balance between core damage frequency (CDF) and conditional containment failure probability (CCFP) can serve as an example of this defense-in-depth concept. We have previously recommended that CDF be elevated to a fundamental safety goal. Let us suppose, for example sake, that our acceptance value on this is 10-4 per reactor year. If that is the value actually achieved by the design, then a CCFP of about 0.5 has been shown (NUREG-1150) to be generally sufficient to meet the safety goal regulatory objective of individual risk of prompt fatality [which can be adequately represented by an acceptance value of 10-5 per reactor year on large, early release frequency (LERF) as noted in Regulatory Guide 1.174]. Does this CCFP provide sufficient defense in depth?

In our view, three acceptance criteria must be satisfied -- one each on CDF, LERF, and the epistemic uncertainty associated with LERF. The Safety Goal Policy Statement suggests candidate acceptance values on CDF and LERF. In addition to these, we must establish the acceptance value on the uncertainty associated with LERF. For the

particular value of LERF achieved, let's say that the acceptance value has been set by policy to be on the epistemic uncertainty that can be directly developed from the PRA [but which properly reflects the unquantified aleatory uncertainties]. Now suppose our PRA uncertainty analysis tells us that the quantified uncertainty for this design is greater than the acceptance value. Employing our concept, the design with the 0.5 CCFP does not have sufficient defense in depth. The design must, then, include provisions for more defense in depth [e.g., a better containment perhaps] or reduction of the LERF to values for which the achieved uncertainty is acceptable. The acceptance value on uncertainty for any given regulatory objective could be a function of the absolute value achieved for the regulatory objective. That is, as the achieved mean value for LERF gets further below the acceptance value, the acceptable level of uncertainty on its determination can be greater.

We believe this concept of defense in depth can provide a rational way to develop sufficiency limits wherever the defense-in-depth measures can be directly evaluated by PRA. We acknowledge however, that considerable judgment will have to be exercised to set limits on uncertainty, especially uncertainties not quantified by the PRA. Our preceding example suggests one approach to managing these uncertainties.

For those regulatory functions that are not well suited for PRA or where the current capabilities of PRAs are not sufficient, we suggest that the limits on application of defense in depth be placed at levels lower than the top-level safety objectives (see Figure 1 of attached paper). We emphasize that, even under these circumstances, the PRA can still dictate when defense in depth is needed. Let us illustrate how we envision defense in depth to be applied under these circumstances with an example. Fire is one of the initiating events of interest. PRAs quantify the occurrence of fires in nuclear power plants and, among other things, their impact on control and power cables. The plant response to the loss of the relevant systems (due to the loss of these cables) is also analyzed.

The frequency of fires in specific critical locations, that is, locations in which cables of redundant systems may be damaged, is estimated in the PRA using experience-based rates of occurrence of fires, multiplied by subjective estimates of the fraction of fires that are large enough to have the potential to cause damage and the fraction of those fires that occur in the specified critical locations. This is a highly subjective part of the risk assessment (therefore, highly uncertain). It is, therefore, a suitable area to invoke defense in depth and to impose prescriptive requirements regarding the prevention of fires in those critical locations [e.g., strict administrative controls and periodic inspections]. Thus, the relative inadequacy of the PRA model suggests how defense in depth should be applied at levels lower than the top-level safety objectives.

We further realize that the fire risk assessment does not include the damaging effects of the smoke generated by a fire. This is a case of omission of a potentially significant effect. Therefore, we would, again, resort to defense in depth and may demand barriers to limit the spread of smoke and to protect sensitive equipment.

Since the impact on the risk metrics of these lower-level defense-in-depth measures cannot be quantified, nor can the uncertainties, the necessity and sufficiency of the

defense-in-depth measures will have to be simply prescribed and that prescription would constitute the acceptance

criteria.

We note that our first example dealing with CDF and CCFP addresses the top level of Figure 1 of the attached paper. If one adopts the structuralist viewpoint at that level, as the paper's preliminary proposal suggests, then the tradeoffs of our example between CDF and CCFP will have to be performed under the assumption that at least some level of defense in depth will be required. If, on the other hand, one adopts the rationalist view even at that level, it is conceivable that the LERF objectives could be satisfied without a containment. Our second example dealing with fires exemplified the rationalist view at lower levels, as the preliminary proposal recommends.

We acknowledge that these preliminary thoughts on the role of defense in depth in a risk-informed regulatory system identify a direction but fall short of closing the issue. We recommend that the Commission give further consideration to this matter.

Sincerely,

/s/

Dana A. Powers
Chairman

References:

1. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
2. U. S. Nuclear Regulatory Commission, NUREG-1150, Vols. 1-3, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," December 1990.
3. Report dated August 15, 1996, from T. S. Kress, Chairman, ACRS, to Shirley A. Jackson, Chairman, NRC, Subject: Risk-Informed, Performance-Based Regulation and Related Matters.
4. Memorandum dated March 5, 1999, from Annette Vietti-Cook, Secretary of the NRC, to John T. Larkins, Executive Director, ACRS, Subject: Staff Requirements - Meeting with the Advisory Committee on Reactor Safeguards, February 3, 1999.

Attachment:

U. S. Nuclear Regulatory Commission, Advisory Committee on Reactor Safeguards, J. N. Sorensen, G. E. Apostolakis, T. S. Kress, D. A. Powers, "On the Role of Defense in Depth in Risk- Informed Regulation," to be presented at PSA 1999, August 22-25, 1999.

ON THE ROLE OF DEFENSE IN DEPTH IN RISK-INFORMED REGULATION

To be presented at PSA '99
Washington, D.C.
August 22-25, 1999

J. N. Sorensen, Senior Fellow
G. E. Apostolakis, Member
T. S. Kress, Member
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Advisory Committee on Reactor Safeguards
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ABSTRACT

The nascent implementation of risk informed regulation in the United States suggests a need for reexamination of the Nuclear Regulatory Commission's (NRC) defense in depth philosophy and its impact on the design, operation, and regulation of nuclear power plants. This reexamination is motivated by two opposing concerns: (1) that the benefits of risk informed regulation might be diminished by arbitrary appeals to defense in depth, and (2) that the implementation of risk informed regulation could undermine the defense in depth philosophy. From either perspective, two questions are suggested: (1) How is defense in depth defined? (2) How should the implementation of risk informed regulation alter our view of defense in depth? A preliminary proposal for the role of defense in depth in a risk- informed regulatory system is presented.

HISTORICAL DEVELOPMENT

Defense in depth is a nuclear industry safety strategy that began to develop in the 1950s. A review of the history of the term indicates that there is no official or preferred definition. Where the term is used, if a definition is needed, one is created consistent with the intended use of the term. Such definitions are often made by example.

In a 1967 statement submitted to the Joint Committee on Atomic Energy by Clifford Beck, then Deputy Director of Regulation for the Atomic Energy Commission, three basic lines of defense for nuclear power reactor facilities were described. The first line was the prevention of accident initiators through superior quality of design, construction and operation. The second line was engineered safety systems designed to prevent mishaps from escalating into major accidents. The third line was consequence-limiting safety systems designed to confine or minimize the escape of fission products to the environment.

A 1969 paper by an internal study group of the Atomic Energy Commission identified the issue of balance among accident prevention, protection, and mitigation, with the conclusion that the greatest emphasis should be put on prevention, the first line of defense.

A 1994 NRC document identifies the elements of the defense in depth safety strategy as accident prevention, safety systems, containment, accident management, and siting and emergency plans. Other interpretations of defense in depth can be found in INSAG-3 and INSAG-10

The historical record indicates an evolution of the term from a narrow application to the multiple barrier concept to an expansive application as an overall safety strategy. The term has increased in scope and gained stature over time. The history also indicates that defense in depth is considered to be a concept, an approach, a principle or a philosophy, as opposed to being a regulatory requirement per se.

Currently the term is commonly used in two different senses. The first is to denote the philosophy of high level lines of defense, such as prevent accident initiators from occurring, terminate accident sequences quickly, and mitigate accidents that are not successfully terminated. The second is to denote the multiple physical barrier approach, most often exemplified by the fuel cladding, primary system, and containment.

One of the essential properties of defense in depth is the concept of successive barriers or levels. This concept applies equally well to multiple physical barriers and to high level lines of defense. A closely related attribute would be requiring a reasonable balance among prevention, protection and mitigation.

EMERGING REGULATORY PRACTICE

The most recent NRC policy statement that deals with defense in depth is the Probabilistic Risk Assessment (PRA) Policy statement published in 1995, which states, in part:

"The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

The policy statement, thus, places PRA in a subsidiary role to defense in depth.

In 1998, the NRC published Regulatory Guide 1.174. This guide establishes an approach to risk-informed decision making, acceptable to the NRC staff, which includes the provision that proposed changes to the current licensing basis must be consistent with the defense in depth philosophy. The RG 1.174 discussion states that, "The defense in depth philosophy . . . has been and continues to be an effective way to account for uncertainties in equipment and human performance." The discussion goes on to say that PRA can be

used to help determine the appropriate extent of defense in depth, which, by example, is equated to balance among core damage prevention, containment failure prevention and consequence mitigation. The regulatory guide thus addresses the concern of preventing risk-informed regulation from undermining defense in depth. Defense in depth is primary, with PRA available to measure how well it has been achieved.

STRUCTURALIST MODEL

We have identified two different schools of thought (models) on the scope and nature of defense in depth. These models came to be labeled "structuralist" and "rationalist."

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. The requirements for defense in depth are derived by repeated application of the question, "What if this barrier or safety feature fails?" The results of that process are documented in the regulations themselves, specifically in Title 10, Code of Federal Regulations. In this model, the necessary and sufficient conditions are those that can be derived from Title 10. It is also a characteristic of this model that balance must be preserved among the high-level lines of defense, e.g., preventing accident initiators, terminating accident sequences quickly, and mitigating accidents that are not successfully terminated. One result is that certain provisions for safety, for example reactor containment and emergency planning, must be made regardless of our assessment of the probability that they may be required. Accident prevention alone is not relied upon to achieve an adequate level of protection.

There does not appear to be any question that the implementation of defense in depth up to the present time reflects the structuralist model. While this philosophy has served the industry well from the safety perspective, it is now realized that, in some instances, it has led to excessive regulatory burden. Furthermore, the lack of an integrated view of the reactor systems has resulted in some significant accident sequences not being identified until PRA was developed, e.g., the interfacing-systems LOCA sequence.

The next issue, then, becomes how should the insights from PRA be integrated into this structure to reduce unnecessary burden and make it more rational? In the structuralist model, defense in depth is primary, with PRA available to measure how well it has been achieved.

THE RATIONALIST MODEL

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 techniques. The process envisioned by the rationalist is: (1) establish quantitative acceptance criteria, such as the quantitative health objectives, core damage frequency and large early release frequency, (2) analyze the system using PRA methods to establish that the acceptance criteria are met, and (3) evaluate the uncertainties in the analysis, especially those due to

model incompleteness, and determine what steps should be taken to compensate for those uncertainties. In this model, the purpose of defense in depth is to increase the degree of confidence in the results of the PRA or other analyses supporting the conclusion that adequate safety has been achieved.

The underlying philosophy here is that the probability of accidents must be acceptably low. Provisions made to achieve sufficiently low accident probabilities are defense in depth. It should be noted that defense in depth may be manifested in safety goals and acceptance criteria which are input to the design process. In choosing goals for core damage frequency and conditional containment failure probability, for example, a judgement is made on the balance between prevention and mitigation.

What distinguishes the rationalist model from the structural model is the degree to which it depends on establishing quantitative acceptance criteria, and then carrying formal analyses, including analysis of uncertainties, as far as the analytical methodology permits. The exercise of engineering judgement, to determine the kind and extent of defense in depth measures, occurs after the capabilities of the analyses have been exhausted.

A PRELIMINARY PROPOSAL

The structuralist and rationalist models are not generally in conflict. Both can be construed as a means of dealing with uncertainty. Neither incorporates any reliable means of determining when the degree of defense in depth achieved is sufficient. In the final analysis, they both depend on knowledgeable people discussing the risks and uncertainties and ultimately agreeing on the provisions that must be made in the name of defense in depth. The fundamental difference is that the structural model accepts defense in depth as the fundamental value, while the rationalist model would place defense in depth in a subsidiary role.

The remaining question is which model provides the better basis for moving forward with risk- informed regulation. How can capricious imposition of defense- in-depth be prevented from undermining the focus that can be provided by risk- informed methods of regulation? PRA methods have identified gaps in the regulations and in the safety profiles of individual plants. They have also identified regulations and plant systems that do not make a significant contribution to safety. Typically, however, regulatory reactions to findings that regulations or plant systems are superfluous to safety have been less aggressive than reactions to apparent safety deficiencies.

Two options can be identified:

(1) Recommend defense in depth as a supplement to risk analysis (the rationalist view)

(2) Recommend a high-level structural view and a low-level rationalist view.

Option (1) requires a significant change in the regulatory structure. The place of defense in depth in the regulatory hierarchy would have to change. The PRA policy statement

could no longer relegate PRA to a position of supporting defense in depth. Defense in depth would become an element of the overall safety analysis.

Option (2) is to a large degree compatible with the current regulatory structure. The structuralist model of defense in depth would be retained as the high-level safety philosophy, but the rationalist model would be used at lower levels in the safety hierarchy. An example is shown in Figure 1.

The PRA uncertainties increase as we move from the initiating events to risk (from left to right). The structuralist view dictates that intermediate goals be set, such as core damage frequency (CDF), large early release frequency (LERF) or conditional containment failure probability (CCFP),or frequency-consequence (F-C) curves. This would satisfy the requirement of balance between prevention and mitigation. We note that the actual numerical value chosen for core damage frequency can express a preference for prevention, and such a preference is unrelated to defense in depth. One could proceed and set goals at the "cornerstone" level, i.e., one level below. This could include goals on initiating- event frequencies, safety-function or safety- system unavailabilities, and so on. How far down one would go would be a policy issue. The structuralist view would not be applied at lower levels.

The rationalist model would be applied at levels lower than the cornerstones of Figure 1. Defense in depth would be used only to address uncertainties in PRA at the lower levels, thus becoming an element of the overall safety analysis. For events or processes that are not modeled in PRA, defense in depth would play its traditional role. Such is the case with the impact of smoke from fires on plant safety. Current fire risk assessments do not account for the effects of smoke, therefore, prescriptive defense-in-depth based measures would be taken to limit this impact.

We view Option (2) as a pragmatic approach to reconciling defense in depth with risk-informed regulation. There can be little doubt, however, that the rationalist model, Option (1), will ultimately provide the strongest theoretical foundation for risk-informed regulation. When more experience has been gained with the application of PRA in the design and regulation of nuclear power plants, when PRA models can adequately treat most of the phenomena of interest, the role of defense in depth can and should be changed to one of supporting the risk analyses. This transition will need to be supported by the development of subsidiary principles from which necessary and sufficient conditions could be derived.

Note

The views expressed in this paper are the authors' and do not necessarily represent the views of the Advisory Committee on Reactor Safeguards

REFERENCES

1. C. Beck, "Basic Goals of Regulatory Review: Major Considerations Affecting Reactor Licensing," Statement submitted to the Joint Committee on Atomic Energy, Congress of the United States, Hearings on Licensing and Regulation of Nuclear Reactors, April 4,5,6,20, and May 3, 1967.
2. Internal Study Group, "Report to the Atomic Energy Commission on the Reactor Licensing Program," submitted to the Joint Committee on Atomic Energy, Congress of the United States, Hearings on AEC Licensing Procedure and Related Legislation, June 1969.
3. F. E. Haskin, and A. L. Camp,, "Perspectives on Reactor Safety," NUREG/CR-6042, Nuclear Regulatory Commission, Washington, DC, March 1994.
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6. U. S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities; Final Policy Statement," Federal Register, 60 FR 42622
7. U. S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Regulatory Guide 1.174, June 1998

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STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE

December 8, 2003

RE: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 263
Extended Power Uprate - State of Vermont Questions

Richard Ennis, Project Manager
U.S. Nuclear Regulatory Commission
Washington, D.C., 20555

Dear Mr. Ennis,

We have received a copy of Entergy Nuclear Vermont Yankee's (Entergy's) request of September 10, 2003, to amend Vermont Yankee Nuclear Power Station's operating license to increase the maximum authorized power level from 1593 megawatts thermal (MWt) to 1912 MWt. Accompanying Entergy's request is a non-proprietary version of NEDC-33090, *Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate ("PUSAR")*, September 2003 (Attachment 6).

We have developed certain preliminary questions from review of the September 10, 2003 request:

1. We note that Entergy's request relies upon a proprietary version of the *Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate ("PUSAR")*, NEDC-33090P, September 2003, which was provided to the NRC as Attachment 4, but which was withheld from public disclosure. In addition, we note that PUSAR relies heavily upon a proprietary document which your agency has approved, GE Nuclear Energy, *Constant Pressure Power Uprate Licensing Topical Report ("CLTR")*, NEDO-33004P-A, July 2003. Your March 31, 2003 approval of CLTR contains proprietary information. Furthermore, it appears the review and approval process of CLTR may depend on earlier proprietary documents, known as ELTR1 and ELTR2, and their related proprietary safety evaluations.

   In order to understand the safety implications of Entergy's proposal, Vermont, through its Department of Public Service, needs to be able to review this proprietary information. Specifically, Vermont needs to be able to review proprietary documents from others upon which NRC will rely in its consideration of the acceptability of Entergy's request, and Vermont needs to receive proprietary requests for additional information, review comments and evaluations that NRC may make based on proprietary documents.

Docket No. 50-271
DPS Exhibit #13
3 Pages

We are willing to enter into necessary confidentiality agreements to allow our needs to be met with regard to this proprietary material. Therefore, we ask that you identify a point of contact with whom we can execute the necessary documentation.

2. We have questions regarding Entergy's request to change its licensing basis to allow crediting of containment pressure for calculating certain pumps net positive suction head (NPSH) following postulated loss-of-coolant accidents (LOCA), station blackouts, and Appendix R fire events:

   a. It appears the base guidance for reviewing this area is Standard Review Plan (SRP) 6.2.2, *Containment Heat Removal Systems*, Rev. 4, October 1985. SRP 6.2.2 appears to follow Regulatory Guide 1.1 (Safety Guide 1) and is unequivocal that credit may not be taken for containment pressurization for NPSH considerations. However, the draft Review Standard for Extended Power Uprates, RS-001, December 2002, indicates that the review standard for this area is SRP 6.2.2, as supplemented by Draft Regulatory Guide (DG) 1107, *Water Sources for Long-term Recirculation Cooling following a Loss-of-Coolant Accident*, February 2003. DG 1107, at 7, includes the statement:

      > Predicted performance of the emergency core cooling and the containment heat removal pumps *should be* independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions... However, for some operating reactors, credit for containment accident pressure *may be necessary*. This should be minimized to the extent possible. [Emphasis added.]

      1) What guidance does the agency have for determining whether "credit for containment accident pressure [is] necessary"?

      2) Does the agency believe that it is *necessary* to operate at extended uprated power level, thereby creating the necessity for allowing credit for containment accident pressure? If the answer is in the affirmative, please identify the reason the agency thinks operating at extened uprated power level is *necessary*?

      3) What is the agency's policy regarding review to draft (rather than final) review guidance?

   b. Regulatory Position 2.1.1.2 of DG 1107 (at 16) states:

      > For certain operating reactors for which the design cannot be *practicably altered*, compliance with Regulatory Position 2.1.1.1 [i.e., no credit for containment accident pressure] may not be possible.

Does the agency consider operation at OLTP to be a practicable alteration to allow compliance with Regulatory Position 2.1.1.1?

c. At what uprated power level could Vermont Yankee operate and not claim credit for containment accident pressure in its NPSH calculations?

d. Could you please identify for which licensees you have found it necessary to allow credit for containment accident pressure, and the reasons you found it necessary?

e. VY PUSAR Table 4-2 and Figure 4-6 identify that containment accident pressure credit is taken for a period over two days after an accident. Since this constitutes the use of the reactor containment in a new manner, i.e., as an engineered safety feature to guarantee a minimum level of pressure over a 50 hour period, is additional containment pressure testing required to demonstrate pressure will be maintained for that period?

f. What is the safety implication if credit for containment accident pressure is allowed? What is the agency's basis for allowing the regulatory requirement changed proposed by DG-1107?

3. In Attachment 7 to License Amendment Request for VY EPU, Entergy provides justification for exception to large transient testing. It does not appear that Entergy discusses the April 16, 2003 inadvertent opening of a power operated relief valve (PORV) at Quad Cities 2 and its role in the second failure of the steam dryer. Should this experience at Quad Cities 2 be considered for the decision whether to large transient testing is required?

4. VY PUSAR Section 4.6 states that VYNPS does not use a Main Steam Isolation Valve Leakage Control System. Why isn't the alternate leakage treatment pathway, described in Entergy's Technical Specification Proposed Change No. 262 (Alternate Source Term), considered a Main Steam Isolation Valve Leakage Control System?

We appreciate your consideration of these items and your assistance in helping us understand the aspects of Vermont Yankee's proposed power uprate. If you have questions about these items, please call me at 802-828-3349.

Sincerely,

William K. Sherman

William K. Sherman
Vermont State Nuclear Engineer

cc: David O'Brien - Commissioner
Ledyard Marsh - NRC
David McElwee - Entergy Nuclear Vermont Yankee

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 29, 2004

Mr. William K. Sherman
Vermont Department of Public Service
112 State Street
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Montpelier, VT 05620-2601

Dear Mr. Sherman:

I am responding to your letter dated December 8, 2003, to the U.S. Nuclear Regulatory Commission (NRC), which provided questions regarding the license amendment request from Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. for the Vermont Yankee Nuclear Power Station (VYNPS) dated September 10, 2003. The proposed license amendment would allow an increase in the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWT) to 1912 MWT.

The NRC staff's response to your questions is enclosed. As you are aware, the NRC's review of the VYNPS power uprate amendment request is not yet complete. As such, we have not reached any conclusions concerning the acceptability of the proposed amendment. We intend to conduct this review in a clear and open manner to ensure participation by interested stakeholders. All comments received, either formally (such as by your letter), or informally (such as at the March 31, 2004 power uprate public meeting in Vernon, Vermont), will be considered by the NRC staff in the course of our review.

We believe that the extensive technical review performed by the NRC staff using our new Review Standard, combined with the inspections prescribed by the reactor oversight process, as enhanced by the new engineering inspection described in our letter to the Vermont Public Service Board dated May 4, 2004, will provide the information necessary for the NRC staff to make a decision on whether VYNPS can operate safely under uprated power conditions. The NRC will not approve the VYNPS power uprate, or any proposed change to a plant license, unless the NRC staff can conclude that the proposed change will be executed in a manner that assures the public's health and safety.

We appreciate your attention to this matter and hope that we have clearly addressed your questions. If you have any further questions, please contact me at 301-415-1420.

Sincerely,

RB Ennis

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

Docket No. 50-271
DPS Exhibit #14
15 Pages

Vermont Yankee Nuclear Power Station

cc:

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RESPONSES TO QUESTIONS FROM

STATE OF VERMONT, DEPARTMENT OF PUBLIC SERVICE

RELATED TO PROPOSED POWER UPRATE

FOR VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

By letter dated September 10, 2003, as supplemented on October 1, 2003, October 28, 2003 (2 letters), January 31, 2004 (2 letters), March 4, 2004, and May 19, 2004, Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee), submitted a license amendment request to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed license amendment would allow an increase in the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWT) to 1912 MWT.

In a letter dated December 8, 2003, the State of Vermont, Department of Public Service (DPS), requested that the NRC respond to questions regarding the proposed power uprate license amendment request for VYNPS. The NRC's responses to the DPS questions are provided below.

Entergy's proposed power uprate license amendment for VYNPS is currently under review by the NRC. The NRC staff has not reached any conclusions concerning the acceptability of the licensee's request at this point in the review. Therefore, the NRC's responses to the DPS questions are answered in generic terms, and do not convey or represent an NRC staff position regarding the proposed amendment.

DPS Question 1

We note that Entergy's request relies upon a proprietary version of the *Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate ("PUSAR")*, NEDC-33090P, September 2003, which was provided to the NRC as Attachment 4, but which was withheld from public disclosure. In addition, we note that PUSAR relies heavily upon a proprietary document which your agency has approved, GE Nuclear Energy, *Constant Pressure Power Uprate Licensing Topical Report ("CLTR")*, NEDO-33004P-A, July 2003. Your March 31, 2003 approval of CLTR contains proprietary information. Furthermore, it appears the review and approval process of CLTR may depend on earlier proprietary documents, known as ELTR1 and ELTR2, and their related proprietary safety evaluations.

In order to understand the safety implications of Entergy's proposal, Vermont, through its Department of Public Service, needs to be able to review this proprietary information. Specifically, Vermont needs to be able to review proprietary documents from others upon which NRC will rely in its consideration of the acceptability of Entergy's request, and Vermont needs to receive proprietary requests for additional information, review comments and evaluations that NRC may make based on proprietary documents.

We are willing to enter into necessary confidentiality agreements to allow our needs to be met with regard to this proprietary material. Therefore, we ask that you identify a point of contact with whom we can execute the necessary documentation.

NRC Response to DPS Question 1

Based on NRC staff discussions with Mr. David McElwee of Entergy, and our previous discussions with you, it is our understanding that you previously entered into non-disclosure agreements with those contractors employed by Entergy that developed proprietary information for the VYNPS power uprate submittal. It is also our understanding that Entergy has provided copies of the documents containing proprietary information to you when requested. Entergy has informed the NRC staff that they are willing to continue that practice during the NRC review process. These agreements should allow you to obtain copies of the documents referenced in your question, including NRC safety evaluations and requests for additional information which contain proprietary information from these Entergy contractors. Mr. McElwee may be reached at 802-258-4112 if you have any further questions regarding the existing non-disclosure agreements.

Although we believe that the practice described above should meet your needs, if you have any difficulty in obtaining any information that you need to fulfill your responsibility to the people of the State of Vermont, please contact the NRC Project Manager, Mr. Richard Ennis, at 301-415-1420.

DPS Question 2.a.1

We have questions regarding Entergy's request to change its licensing basis to allow crediting of containment pressure for calculating certain pumps net positive suction head (NPSH) following postulated loss-of-coolant accidents (LOCA), station blackout, and Appendix R fire events:

a. It appears the base guidance for reviewing this area is Standard Review Plan (SRP) 6.2.2, *Containment Heat Removal Systems*, Rev. 4, October 1985. SRP 6.2.2 appears to follow Regulatory Guide 1.1 (Safety Guide 1) and is unequivocal that credit may not be taken for containment pressurization for NPSH considerations. However, the draft Review Standard for Extended Power Uprates, RS-001, December 2002, indicates that the review standard for this area is SRP 6.2.2, as supplemented by Draft Regulatory Guide (DG) 1107, *Water Sources for Long-term Recirculation Cooling following a Loss-of-Coolant Accident*, February 2003. DG 1107, at 7, includes the statement:

> Predicted performance of the emergency core cooling and the containment heat removal pumps *should be* independent of the calculated increases in containment pressure caused by postulated LOCAs in order to ensure reliable operation under a variety of possible accident conditions...However, for some operating reactors, credit for containment pressure *may be necessary*. This should be minimized to the extent possible. [Emphasis added.]

1) What guidance does the agency have for determining whether "credit for containment accident pressure [is] necessary"?

NRC Response to DPS Question 2.a.1

The NRC has allowed credit for containment accident pressure in calculating the available NPSH of the emergency core cooling system (ECCS) and containment heat removal pumps in some boiling water reactors (BWRs) and, in fewer cases, in pressurized water reactors (PWRs). Using conservative design basis assumptions, some licensees have credited containment accident pressure for NPSH calculations, when the existing plant design cannot be practicably altered (e.g., replacement of ECCS pumps) in order to maintain the calculated available NPSH greater than the required NPSH. The licensee's determination that the design cannot be practicably altered is what was intended by the statement in DG 1107 that "credit for containment accident pressure may be necessary."

The NRC staff allows such credit to be taken only when the licensee's analyses and justification for the proposed licensing basis change demonstrate that there is reasonable assurance that the credited pressure will exist for the events (e.g., postulated design-basis accidents (DBAs), station blackout, Appendix R postulated fires, anticipated transients without scram) and time period for which the credit is required. Ensuring containment integrity and avoiding overcooling of the containment due to excessive use of containment sprays are key considerations in determining whether the credited pressure will be available during the required time period.

The NRC's guidance regarding whether it is acceptable to credit containment accident pressure has evolved over the years. The current guidance is contained in Regulatory Guide (RG) 1.82, Revision 3, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident" dated November 2003, which has replaced DG 1107. This RG describes methods acceptable to the NRC staff for implementing requirements with respect to the sumps and suppression pools performing the functions of water sources for emergency core cooling, containment heat removal, or containment atmosphere clean up. Methods and solutions different from those set out in the RG are acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

RG 1.82, Revision 3, page 2 states, in part, that:

> This regulatory guide has also been revised to include guidance previously provided in Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps." The provisions of Regulatory Guide 1.1 have been updated in this guide to reflect the results of the NRC's review of responses to Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," dated October 7, 1997.

Based on review of your question, the NRC staff has discussed the need for more clarity in the guidance on credit for containment accident pressure. We realize the fact that we did not formally withdraw RG 1.1, which did not allow credit for containment accident pressure, has led to confusion about our technical position. We will both formally withdraw RG 1.1 since it has been superceded, and revise RG 1.82 to more clearly explain how credit for containment accident pressure can be found acceptable. In addition, the staff plans to update SRP 6.2.2 to reference the latest revision of RG 1.82, and to delete references to RG 1.1.

Attachment 1 is provided in order to explain the NRC's current position and the evolution of the NRC's guidance regarding credit for containment accident pressure.

DPS Question 2.a.2

Does the agency believe that it is *necessary* to operate at extended uprated power level, thereby creating the necessity for allowing credit for containment accident pressure? If the answer is in the affirmative, please identify the reason the agency thinks operating at extended uprated power level is *necessary*?

NRC Response to DPS Question 2.a.2

The NRC staff makes no judgment on whether a proposed license amendment, such as a power uprate request, is necessary as long as the proposed changes satisfy NRC requirements and ensure safe operation of the facility. Some power uprate requests create a necessity for licensees to take credit for containment accident pressure. However, the NRC will allow this credit to be taken only if there is reasonable assurance that safety will be maintained. As discussed in the response to question 2.a.1, the NRC staff will revise RG 1.82 to more clearly explain the conditions under which this credit can be taken.

DPS Question 2.a.3

What is the agency's policy regarding review to draft (rather than final) review guidance?

NRC Response to DPS Question 2.a.3

RGs are issued to describe and make available to the public such information as methods acceptable to the staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and guidance to applicants. RGs are not substitutes for regulations, and compliance with RGs is not required. RGs are issued in draft form for public comment to involve the public in developing the regulatory positions. DGs are subject to further revision as a result of public comment or further staff review; they, therefore, do not represent official NRC staff positions.

Although Review Standard RS-001 references DG 1107, this draft guidance has now been replaced by RG 1.82, Revision 3, which will be used for the VYNPS power uprate amendment review (i.e., current guidance is no longer draft guidance). The regulatory positions contained in RG 1.82, Revision 3, regarding NPSH of ECCS and containment heat removal pumps (Section C.1.3.1 for PWRs and C.2.1.1 for BWRs) were revised slightly from the same sections in DG 1107. However, the basic staff positions remained unchanged.

DPS Questions 2.b and 2.c

(b) Regulatory Position 2.1.1.2 of DG 1107 (at 16) states:

> For certain operating reactors for which the design cannot be *practicably altered*, compliance with Regulatory Position 2.1.1.1 [i.e., no credit for containment accident pressure] may not be possible.
>
> Does the agency consider operation at OLTP [original licensed thermal power] to be a practicable alteration to allow compliance with Regulatory Position 2.1.1.1?

(c) At what uprated power level could Vermont Yankee operate and not claim credit for containment accident pressure in its NPSH calculations?

NRC Response to DPS Questions 2.b and 2.c

Our understanding of the meaning of your question 2.b is whether the NRC staff should consider not evaluating power uprate requests that include a request for containment accident pressure credit in order to meet the intent of Regulatory Position C.2.1.1.2 in RG 1.82 (formerly DG 1107). RGs describe methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and guidance to applicants. The intent of Regulatory Position C.2.1.1.2 was to provide guidance to licensees on considerations for calculating the available NPSH if they determine that the existing plant design cannot be practicably altered (e.g., replacement of ECCS pumps) in order to maintain the available NPSH greater than the required NPSH. Regulatory Position C.2.1.1.2 was not intended to preclude a licensee from requesting a power uprate that includes a request for containment accident pressure credit.

With respect to question 2.c, the NRC staff has not performed calculations to determine the power at which credit for containment pressure is not required when using conservative assumptions and the licensee has not presented such an analysis to us.

DPS Question 2.d

Could you please identify for which licensees you have found it necessary to allow credit for containment accident pressure, and the reasons you found it necessary?

NRC Response to DPS Question 2.d

The NRC does not maintain a list of plants for which credit for containment accident pressure has been approved, but the following list is believed to be reasonably complete.

Beaver Valley Unit 1 (PWR)
Browns Ferry Units 2 and 3 (BWR Mark I Containment)
Brunswick Units 1 and 2 (BWR Mark I Containment)
Cooper (BWR Mark I Containment)
Dresden Units 2 and 3 (BWR Mark I Containment)
Duane Arnold (BWR Mark I Containment)
FitzPatrick (BWR Mark I Containment)
Fort Calhoun (PWR)
Hatch Units 1 and 2 (BWR Mark I Containment)
Monticello (BWR Mark I Containment)
North Anna Units 1 and 2 (PWRs)
Oconee Units 1, 2 and 3 (PWRs)
Oyster Creek (BWR Mark I Containment)
Peach Bottom Units 2 and 3 (BWR Mark I Containment)
Pilgrim (BWR Mark I Containment)
Quad Cities Units 1 and 2 (BWR Mark I Containment)
Surry Units 1 and 2 (PWRs)

As previously discussed in the answer to question 2.a.1, some licensees have credited containment accident pressure for NPSH calculations when the existing plant design cannot be practicably altered in order to maintain the available NPSH greater than the required NPSH. The NRC staff will allow such credit to be taken only when the licensee's analyses and justification for the proposed licensing basis change demonstrate that there is reasonable assurance that the credited pressure will exist for the events and time period for which the credit is required. As discussed in Attachment 1, an increase in licensed power level is not the context for crediting containment accident pressure in all cases.

DPS Question 2.e

VY PUSAR Table 4-2 and Figure 4-6 identify that containment accident pressure credit is taken for a period over two days after an accident. Since this constitutes the use of the reactor containment in a new manner, i.e., as an engineered safety feature to guarantee a minimum level of pressure over a 50 hour period, is additional containment pressure testing required to demonstrate pressure will be maintained for that period?

NRC Response to DPS Question 2.e

The VYNPS reactor containment already serves as an engineered safety feature. It serves as a pressure barrier to minimize leakage. Tests are done, as specified in the VYNPS Technical Specifications (TSs), in compliance with Title 10 of the *Code of Federal Regulations* Part 50, Appendix J, to ensure the pressure retaining capability of the containment. These tests verify compliance with a stringent leakage rate limit. As discussed in Attachment 1, the containment integrity is continuously monitored in the control room. In addition, the maximum TS-allowable containment leakage is typically assumed in the calculation of the available NPSH (i.e., NPSH is determined consistent with the plant licensing basis). This assumption will be verified as part of the NRC staff's review of the proposed power uprate request. Based on the above considerations, additional containment pressure testing is not considered necessary.

DPS Question 2.f

What is the safety implication if credit for containment accident pressure is allowed? What is the agency's basis for allowing the regulatory requirement change[ ] proposed by DG 1107?

NRC Response to DPS Question 2.f

As discussed in the response to question 2.a.1, the NRC staff allows containment accident pressure credit to be taken only when the licensee's analyses and justification for the proposed licensing basis change demonstrate that there is reasonable assurance that the credited pressure will exist for the events and time period for which the credit is required. This provides assurance that the ECCS and containment heat removal pumps will have adequate NPSH to perform their intended safety functions. As part of the review for the proposed VYNPS extended power uprate (EPU), the NRC has requested the licensee to provide additional information to further justify relying on containment accident pressure for ECCS pump NPSH. The request includes having the licensee provide information to address this proposed change from a risk perspective (e.g., potential impact on core-damage frequency). This information is expected to help in the NRC's decision making process to determine if there is reasonable assurance of continued adequate protection of public health and safety if the proposed change is approved.

RG 1.82, Revision 3 (formerly DG 1107) does not contain regulatory requirements. As discussed in the response to question 2.a.3, RGs are issued to describe and make available to the public such information as methods acceptable to the staff for implementing specific parts of the NRC's regulations, techniques used by the staff in evaluating specific problems or postulated accidents, and guidance to applicants. RGs are not substitutes for regulations, and compliance with RGs is not required. As discussed in the response to question 2.a.1, the NRC's position regarding whether it is acceptable to credit containment accident pressure has evolved over the years. In order to understand the NRC's current position and the basis for that position, Attachment 1 provides further details on this issue.

DPS Question 3

In Attachment 7 to License Amendment Request for VY EPU, Entergy provides justification for exception to large transient testing. It does not appear that Entergy discusses the April 16, 2003 inadvertent opening of a power operated relief valve (PORV) at Quad Cities 2 and its role in the second failure of the steam dryer. Should this experience at Quad Cities 2 be considered for the decision whether [ ] large transient testing is required?

NRC Response to DPS Question 3

The recent and emerging issues concerning steam dryer integrity are being evaluated by the NRC staff and are being considered in the review of the VYNPS power uprate amendment request. The NRC staff has requested additional information of the licensee, regarding their proposed exception to large transient testing, to further justify operation at EPU conditions based on the industry experience relative to steam dryer failures.

DPS Question 4

VY PUSAR Section 4.6 states that VYNPS does not use a Main Steam Isolation Valve Leakage Control System. Why isn't the alternate leakage pathway, described in Entergy's Technical Specification Proposed Change No. 262 (Alternate Source Term), considered a Main Steam Isolation Valve Leakage Control System?

NRC Response to DPS Question 4

The term "Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)" refers to a supplemental system that some plants installed as recommended by RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," Revision 1, dated June 1976. As discussed in the NRC's Safety Evaluation for General Electric Topical Report NEDC-31858P dated March 3, 1999 (ADAMS Accession No. ML010640286), to meet this RG, many licensees installed a safety-related MSIV LCS that is designed to eliminate or minimize the direct release of fission products through the MSIVs following a design-basis LOCA. This is usually accomplished by developing a negative pressure in sections of the main steam lines between the MSIVs. In general, this is accomplished by a series of blowers that discharge the MSIV leakage to the Standby Gas Treatment System where it is released. A few plants may have a positive pressure LCS in the main steam lines between the MSIVs. At these plants, MSIV leakage is directed back into containment such that there is no containment bypass leakage through the MSIVs. RG 1.96 discusses the design considerations for a MSIV LCS, including recommendations regarding instrumentation, controls, and interlocks.

The alternate leakage treatment (ALT) pathway, described in Entergy's license amendment request for implementation of an alternate source term at VYNPS, uses the main steam drain line to direct MSIV leakage to the main condenser. The ALT pathway takes advantage of the large volume of the main steam piping and condenser to provide holdup and plate-out of fission products that may leak through closed MSIVs. The ALT pathway method does not utilize instrumentation, controls, interlocks, or equipment such as blowers. Since the term MSIV LCS has specific connotations based on RG 1.96, the ALT pathway is not considered a MSIV LCS.

ATTACHMENT 1
DISCUSSION REGARDING CREDIT FOR CONTAINMENT ACCIDENT PRESSURE

On November 2, 1970, the Atomic Energy Commission issued Regulatory Guide (RG) 1.1 (Safety Guide 1), "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The regulatory position in the RG stated that:

> 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 no increase in containment pressure from that present prior to postulated loss of coolant accidents.

Reactors licensed after issuance of RG 1.1 generally met this guidance.

On December 3, 1985, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA [loss-of-coolant accident] Recirculation Capability Due to Insulation Debris Blockage." This GL discussed the findings related to the resolution of NRC Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The technical findings of USI A-43 are documented in NRC report NUREG-0897, Revision 1, "Containment Emergency Sump Performance, Technical Findings Related to Unresolved Safety Issue A-43," which was issued in October 1985. Although USI A-43 was formulated considering pressurized water reactor (PWR) sumps, the generic concerns applied to both boiling water reactors (BWRs) and PWRs. Therefore, both BWRs and PWRs were considered in NUREG-0897.

The NRC staff's technical findings regarding the resolution of USI A-43 included the following main points, as discussed in GL 85-22: (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. As discussed in the GL, the regulatory analysis for this issue did not support a generic backfit action and resulted in the decision that the revised regulatory guidance would not be applied to plants licensed to operate or that were under construction at the time the GL was issued. GL 85-22 recommended that the revised guidance developed as a result of this issue be used by licensees for any future modifications to thermal insulation installed on primary coolant system piping and components.

As part of the resolution of USI A-43, Standard Review Plan (SRP) 6.2.2 was revised in October 1985 (Revision 4) to include the following acceptance criteria regarding NPSH in the recirculation phase of operation:

> The NPSH analysis will be acceptable if (1) it is done in accordance to the guidance of Regulatory Guide 1.82, Rev. 1 and (2) it is done in accordance with the guidelines of Regulatory Guide 1.1, i.e., is based on maximum expected temperature of the pumped fluid and with atmospheric pressure in containment.

Thus, even after this first examination of the effects of LOCA-generated debris on the available NPSH of emergency core cooling system (ECCS) pumps, the criterion for calculating available NPSH remained that of 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 per square inch gauge which resulted in dislodging

mineral wool insulation, which subsequently blocked emergency pump suction strainers. This event was discussed in NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," dated May 6, 1996.

All BWRs, including Vermont Yankee Nuclear Power Station (VYNPS), met the recommendations of Bulletin 96-03, by the installation of larger, better designed ECCS suction strainers. The design of these strainers took into account plant-specific suction strainer loadings of several types of materials including LOCA-generated debris from dislodged thermal insulation, dislodged paint chips and rust accumulated in the suppression pool which would become thoroughly mixed in the suppression pool water by the turbulence generated by a LOCA. In general, these loadings were predicted to be much higher than anticipated prior to the research which followed the Barsebäck event. This resulted in an increase in the predicted flow resistance across the strainers which resulted in a decrease in calculated available NPSH. In some cases, licensees credited containment accident pressure to meet NPSH requirements for the existing pumps. This was not true for VYNPS. The improved suction strainers were installed at VYNPS during the 1998 refueling outage as discussed in a letter from the licensee dated December 29, 1999 (ADAMS Accession No. ML003671163).

As a related issue, in 1996 and 1997, as a result of NRC inspections, licensee notifications, and licensee event reports, the NRC staff became aware that the available NPSH for ECCS and containment heat removal system pumps may not have been adequate in all cases. This applied to both PWRs and BWRs. In order to understand the extent of the problem, the NRC issued GL 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," on October 7, 1997. This GL requested licensees to provide information necessary to confirm the adequacy of the NPSH available for the ECCS and containment heat removal pumps.

There are no review criteria in GL 97-04 itself. GL 97-04 was a request for information. Specifically, there was no criterion prohibiting the use of containment accident pressure in the calculation of available NPSH in GL 97-04.

In response to GL 97-04, licensees, in some cases, revised their NPSH analyses. Some of the licensees that revised their NPSH analyses proposed credit for containment accident pressure in the calculation of NPSH. The NRC reviewed all responses to GL 97-04 to have reasonable assurance that safety would be maintained. The NRC staff formulated and applied acceptance criteria for these reviews and included the criteria in Draft Regulatory Guide (DG) 1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant Accident" (ADAMS Accession No. ML030550431). Including regulatory positions on NPSH in this DG provided one reference for all regulatory positions related to pump suction issues (vortexing, air entrainment, debris blockage as well as NPSH). DG 1107 was finalized and published as RG 1.82, Revision 3 in November 2003.

As discussed in RG 1.82, Revision 3, Regulatory Position C.1.3.1.2 (for PWRs) and Regulatory Position C.2.1.1.2 (for BWRs), for certain operating plants for which the design cannot be practicably altered, credit for containment accident pressure may be necessary. However, the NRC will allow this credit to be taken only if there is reasonable assurance that safety will be maintained. The NRC has made the judgment that, in these cases, the impact of replacing existing ECCS or containment heat removal pumps with pumps that do not require this credit is not justified based on the design-basis safety analyses done by each plant.

This judgment is based on several factors. The calculated containment accident pressure for determining available NPSH is calculated in a way that underestimates this pressure. For example, the operation of the containment sprays is assumed even though they are not safety-related and, therefore, would not normally be credited in a safety analysis. Operation of the containment sprays significantly reduces the containment pressure which, in this case, is conservative. Credit is also taken for the transfer of heat from the containment atmosphere to various structures to further reduce the calculated containment pressure. Leakage from the containment at the Technical Specification (TS) limit, L_a, is also assumed. The NPSH calculations also overestimate the temperature of the suppression pool water, an important factor in NPSH calculations. The ultimate heat sink temperature is assumed to be at the maximum value allowed by TSs. This limits the amount of heat which can be transferred from the suppression pool and maximizes the suppression pool temperature. Also, the heat transfer capability of the suppression pool cooling system heat exchanger is underestimated.

The rationale for not crediting containment accident pressure, according to RG 1.1, is the possibility of "impaired containment integrity" or excessive operation of the heat removal systems (sprays) resulting in a pressure less than that needed to maintain an adequate NPSH margin.

The primary containment at VYNPS is a Mark I design. The design consists of a drywell which encloses the reactor vessel, a pressure suppression chamber (torus) which stores a large volume of water, a connecting vent system between the drywell and the suppression chamber, isolation valves, containment cooling systems, and other service equipment. During normal operation the containment is inerted, that is, air is removed and the containment is filled with nitrogen gas. A differential pressure is maintained between the drywell and suppression chamber in accordance with the VYNPS TSs. The TSs also limit the maximum containment oxygen concentration.

Instrumentation is provided in the control room to continuously monitor containment integrity. Indications of a degradation of containment integrity from this instrumentation include: a reduction in drywell pressure; a reduction in the drywell to suppression chamber differential pressure, or an increase in the oxygen concentration. Indication of a degradation of containment integrity would prompt appropriate action by the control room operators as required by the VYNPS TSs. In addition, tests are done, as specified in the VYNPS TSs, in compliance with Title 10 of the *Code of Federal Regulations* Part 50, Appendix J, to ensure the pressure retaining capability of the containment. As discussed above, the available NPSH is determined assuming the maximum TS-allowable containment leakage.

The present VYNPS ECCS and containment licensing basis, as with all licensed nuclear power plants, is derived from DBA analyses. The determination of available NPSH is based on design-basis analysis. DBAs are accidents postulated to establish limits on operating conditions and safety-related equipment requirements given in the TSs. The assumptions used in design-basis analyses are chosen to reasonably bound expected conditions. Thus, as explained above, flows, temperatures, pressures, power, etc., bound the expected conditions at which available NPSH is important to safety.

If realistic, rather than conservative and bounding assumptions were used in the design-basis safety analyses, credit for containment accident pressure might not be necessary and, in any case, the containment accident pressure required for available NPSH would be much less than predicted under the conservative design basis assumptions.

The NRC staff has also considered the impact of credit for containment accident pressure for other events during which the ECCS or the containment heat removal system may be called upon to function. For station blackout, anticipated transients without scram and Appendix R postulated fires, the suppression pool conditions are typically less severe than those for the design-basis LOCA. For these postulated events, debris generated are expected to be less than in a LOCA and, therefore, the flow losses are lower for these events than for the design-basis LOCA and the available NPSH consequently greater. For these events, containment accident pressure may not need to be credited. For cases where it is credited, the amount of containment accident pressure is typically less than that credited for the postulated LOCA.

The NRC staff will allow credit for containment accident pressure to be taken only when the licensee's analyses and justification for the proposed licensing basis change demonstrate that there is reasonable assurance that the credited pressure will exist for the events and time period for which the credit is required. Ensuring containment integrity and avoiding overcooling of the containment due to excessive use of containment sprays are key considerations in determining whether the credited pressure will be available during the required time period.

The NRC staff has discussed the need for more clarity in the guidance on credit for containment accident pressure. We realize the fact that we did not formally withdraw RG 1.1, which did not allow credit for containment accident pressure, has led to confusion about our technical position. We will both formally withdraw RG 1.1 since it has been superceded, and revise RG 1.82 to more clearly explain how credit for containment accident pressure can be found acceptable. In addition, the staff plans to update SRP 6.2.2 to reference the latest revision of RG 1.82, and to delete references to RG 1.1.

As previously noted, the NRC staff has not reached any conclusions concerning the acceptability of the VYNPS power uprate request at this point in the review.

STATE OF VERMONT
PUBLIC SERVICE BOARD

Petition of Entergy Nuclear Vermont Yankee and Entergy Nuclear Operations, Inc., pursuant to 30 V.S.A. §248, for a Certificate of Public Good to modify certain generation facilities	))))	Docket No. 6812

Prefiled Direct Testimony of

William Sherman

on behalf of the

Vermont Department of Public Service

May 9, 2003

Summary: Mr. Sherman summarizes his review of the proposed power uprate and presents the Department's conclusions.

Docket No. 50-271
DPS Exhibit #15
2 Pages

Q. Please provide a brief summary of your review.

A. Since VYNPS already exists and since the investment risk associated with the modifications for power uprate would be borne by Entergy and not ratepayers, there are few impacts. I have specific comments regarding the need and economic benefit criteria, 30 V.S.A. §248 (b)(2) and 30 V.S.A. §248 (b)(4).

Q. What are your comments on the economic benefit criterion, 30 V.S.A. §248 (b)(4)?

A. The most common manner to show an economic benefit is to demonstrate that Vermont ratepayers will receive additional power at a favorable price. Entergy is required to offer Vermont Yankee Nuclear Power Corporation (VYNPC) a commercially reasonable opportunity to negotiate for uprate power, but has not yet extended such an offer. Using historical capacity factors and refueling outage durations, it appears that the cost to Entergy per megawatt-hour of uprate power is in the order of $20/MWh or 2.0 cents per kWh.. Therefore, it appears Entergy has the ability to provide Vermont ratepayers with at least a portion of uprate power at a favorable price.

Q. In the answer above you state that Entergy must offer VYNPC the opportunity to negotiate. If the negotiation is with VYNPC, how would benefits of favorable priced uprate power flow to Vermont ratepayers?

A. Vermont utilities, Central Vermont Public Service (CVPS) and Green Mountain Power (GMP) together are the major owners of VYNPC. Power purchased by VYNPC flows through

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VY CALCULATON TITLE PAGE

VYC-808	6	NA	NA
VY Calculation Number	Revision #	Vendor Calculation #	Revision #

Title: Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin following a Loss of Coolant Accident With Fibrous Debris on the Intake Strainers

QA Status: ☒ SC ☐ NNS ☐ OQA Operating Cycle Number* NA

Calculation Supports A Design Change/Specification? ☐ Yes ☒ No NA VYDC/MM/TM/Spec. No.

Implementation Required? ☐ Yes ☒ No

Calculation Done as a Study Only? ☐ Yes ☒ No

Safety Evaluation Number: N/A

Superseded Calculation Number, Title, Revision: (1) VYC-808, Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin following a Loss of Coolant Accident With Fibrous Debris on the Intake Strainers, Revision 5, (2) VYC-1751, Core Spray and Residual Heat Removal Pump Strainer Loss following a Loss of Coolant Accident with Fibrous Debris on the New Intake Strainer, Revision 0. (3) VYC-1670, Limiting Debris Split Fraction for RHR + CS Suction Strainer Design, Rev. 0.

CCF 2/13/01 BCS 2/13/01

For Revisions: List CCns, IIJs, or SAs incorporated/superseded by this revision: CCN1, CCN2, CCN3, and VYS 60/97

Computer Code(s): None

Review and Approval: (Print and Sign Name)

Preparer: B. C. Slifer BC Slifer Date: 9/14/00

Interdiscipline Reviewer(s): NA Date: NA

Independent Reviewer(s): W. TIMOFEEV [signature] Date: 12/13/00

Approved: [signature] CARL D. FAGO Date: 2/13/01

Installation Verification / Final Turnover to DCC:

Open Items Associated with Calculation? ☐ Yes ☒ No ☐ Closed (Section 2.3.2)

Total No. Pages in Package (including all attachments) 78

Installation Verification (Section 2.3.4)

☒ Calculation accurately reflects plant as-built configuration, OR

☐ N/A, calculation does not affect plant configuration

Resolution of documents identified in the Design Output Documents Section of VYAPF 0017.07 initiated (Section 2.3.6)

Carl D. Fago / [signature] / 2/13/01

Printed Name / Signature / Date

Page 1 of ~~26~~ 27 BCS 2/13/01 Pages (body of calculation)

Docket No. 50-271
DPS Exhibit #16
78 Pages

VY CALCULATION DATABASE INPUT FORM

VYC-808	6	NA	NA
VY Calculation/CCN #	Revision #	Vendor Calculation #	Revision #

Vendor Name: NA

PO Number: NA

Originating Department: Fluid Systems

Critical References Impacted: ☐ FSAR ☐ DBD ☐ Reload "Check" the appropriate box if any critical document is identified in the tables below.

EMPAC Asset/Equipment ID Number(s): P-10-1A, P-10-1B, P-10-1C, P-10-1D, P-46-1A, P-46-1B

EMPAC Asset/System ID Number(s): 10, 14

Keywords: Residual Heat Removal (RHR), Core Spray (CS), Net Positive Suction Head (NPSH), Strainer, Pump

For Revision/CCN only: Are deletions to General References, Design Input Documents or Design Output Documents required?

☒ Yes† ☐ No

General References

*Reference #	** DOC #	REV #	*** Reference Title (including Date, if applicable) (See App. A, Section 3.2.7 for Guidance)	**** Affected Program	Critical Reference (√)
1			American National Standard for Centrifugal Pumps, ANSI/HI 1.1-1.5-1994.		
16	BVY 97-138	N/A	Letter, VYNPC to USNRC, "Reply to Inspection Report No. 50-271/97-201," (Appendix B, IFI 97-201-04), dated 10/27/97.		
(c) Attachments 6 and 7			Technical Paper No. 410, "Flow of Fluids Through Valves Fittings and Pipe," Crane Co., 24th Printing, 1988.		
(l) Attachment 6 and (m) Attachment 7			Hydraulic Institute Engineering Data Book, 1st Edition		
(k) Attachment 7			Handbook of Hydraulic Resistance, I. E. Idelchik, 2nd Edition, Hemisphere Publishing Co., New York, 1986.		

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Design Input Documents – The following documents provide design input to this calculation. (Refer to Appendix A, section 4)

*Reference #	** DOC #	REV #	*** Reference Title (including Date, if applicable) (See App. A, Section 3.2.7 for Guidance)	**** Affected Program	Critical Reference (√)
5	VYC-1628F	0	Limiting Torus Temperature Response, Million Second Run		
6	VYC-1919	0	DE&S Calc:DC-A34600-01, "RHR and CS Suction Strainer Assembly Clean Strainer Head Loss"		
2	VYC-1924	0	DE&S Calc:DC-A43600.006, "Vermont Yankee ECCS Suction Strainer Head Loss Performance Assessment, RHR and CS Debris Head Loss Calculation"		

Design Output Documents – **This calculation provides output to the following documents.** (Refer to Appendix A, section 5)

*Reference #	** DOC #	REV #	Document Title (including Date, if applicable)	**** Affected Program	Critical Reference (√)
3	VYC-1677	0	Debris Source Terms for Sizing of Replacement RHR and CS Strainers		

* Reference # - Assigned by preparer to identify the reference in the body of the calculation.
** Doc # - Identifying number on the document, if any (e.g. 5920-0264, G191172, VY6-1286)
*** Reference Title - List the specific documentation in this column. "See attached list" is not acceptable. Design Input/Output Documents should identify the specific design input document used in the calculation or the specific document affected by the calculation and not simply reference the document (e.g., VYDC, MM) that the calculation was written to support.
**** Affected Program - List the affected program or the program that reference is related to or part of. If the reference is FSAR, DBD or Reload (IASD or OPL), check Critical Reference column and check FSAR, DBD or Reload, as appropriate, on this form (above).
† If "yes", attach a copy of "VY Calculation Data" marked-up to reflect deletion (See Section 3.1.8 for Revision and 5.2.3.18 for CCNs).

P.3

VY Calculation Data

[redacted]

☑ CCNs

VY Calculation Number

Vendor Calculation Numbe

Rev No.	Revision Title	Rev Dat	Preparer	Rev Status	Microfilm
0	NPSH MARGIN FOR RHR + CORE SPRAY PUMPS AT DESIGN FLOW RATE FIBROUS INSULATION DEBRIS ON INTAKE STRAINERS FROM A DESIGN BASIS LOCA	03/31/1989	L. TREMBLAY	Superseded	1477-1902
1	NPSH MARGIN FOR RESIDUAL HEAT REMOVAL RHR + CORE SPRAY PUMPS AT DESIGN FLOW RATE INCLUDING FIBROUS INSULATION DEBRIS ON INTAKE STRAINERS FROM A DESIGN BASIS LOCA	10/19/1993	R SWENSON	Superseded	1835-0657
2	NPSH MARGIN FOR RHR RESIDUAL HEAT REMOVAL + CORE SPRAY PUMPS AT DESIGN FLOW RATE INCLUDING FIBROUS INSULATION DEBRIS ON INTAKE STRAINERS FROM A DESIGN BASIS LOCA	12/06/1995	R SWENSON	Superseded	2011-0423
3	CORE SPRAY AND RESIDUAL HEAT REMOVAL PUMP NET POSITIVE SUCTION HEAD MARGIN FOLLOWING A LOSS OF COOLANT ACCIDENT WITH FIBROUS DEBRIS ON THE INTAKE STRAINER	01/13/1998	B SLIFER	Superseded	DCC
4	CORE SPRAY + RESIDUAL HEAT REMOVAL RHR PUMP NET POSITIVE SUCTION HEAD MARGIN FOLLOWING A LOSS OF COOLANT ACCIDENT WITH FIBROUS DEBRIS ON THE INTAKE STRAINERS	05/14/1998	B SLIFER	Superseded	2427-0209
5	CORE SPRAY + RESIDUAL HEAT REMOVAL RHR PUMP NET POSITIVE SUCTION HEAD MARGIN FOLLOWING A LOSS OF COOLANT ACCIDENT LOCA WITH FIBROUS DEBRIS ON THE INTAKE STRAINERS	07/31/1998	B SLIFER	Active	2427-0300
6	CORE SPRAY + RESIDUAL HEAT REMOVAL RHR PUMP NET POSITIVE SUCTION HEAD MARGIN FOLLOWING A LOSS OF COOLANT ACCIDENT LOCA WITH FIBROUS DEBRIS ON THE INTAKE STRAINERS	08/22/2000	B SLIFER	Pending	null

QA Status ☑ SC ☐ NNS ☐ OQA

Operating Cycle No

Supports a Design Change/Spec? ☐

Done as a Study Only? ☐

Related Design Changes:
EDCR 85-01
DCR 98-005

Safety Evaluation:

DBD: ☑

Superseded Calculations: VYC-1685

Superseded by:

Open Items Associated with Calculation ☐

Vender Name: VY

Originating Department: Fluid Systems

Equipment:

P-10-1A
P-10-1B
P-10-1C
P-10-1D
P-46-1A
P-46-1B
P10-1A-D (RHR PUMPS)
P46-1A,B (CS PUMPS)

Systems:

10 Residual Heat Removal System RHR
14 Core Spray System CS

Keywords:

VY Calculation Data

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[redacted]

VY Calculation Number

Vendor Calculation Numbe

☑ CCNs

PSH
RESIDUAL
HEAT
CORE
SPRAY
PUMP
NEGATIVE
SUCTION
HEAD
MARGIN
FLOW
INSULATION
DEBRIS
INTAKE
STRAINER
COOLANT
E7014
CHEMICAL
VOLUME
EMERGENCY
COOLING
NET
POSITIVE
REMOVAL
FIBROUS
CS
RHR
NET POSITIVE SUCTION HEAD
STRAINER
LOCA

eral References:

DMO 96-05
BVY 97-138
CSN 1034
CSN 4805
CSN 5268
CSN 5610
Document E12.5.522 Dated 5/26/98
GE Data Sh Doc 22A1440AB
GE Data Sh Doc 22A2806AB
HANDBOOK OF HYDRAULIC RESISTANCE I E IDELCHIK 2 ED
HYDRAULIC INST. ENG DATA BOOK 1ST ED
NVY 97-130
NVY 98-043
TECH PAPER NO 410 CRANE CO 24TH PRINTING 1988
THSAG VY 98-064
VYS 95-109

Input Documents:

5920-9209 PART 3	4
5920-9284 PART 5	3
Drawing 6202-1	1
FSAR — Delete	14
1168	38
G-191172	57
G-191207	10
G-191211	16

[redacted]

VY Calculation Data

q.6

☑ CCNs

Calculation Number		Vendor Calculation Numbe
ter Sulzer Bingham Pumps to D Yasi 3/26/1999	Delete	
Memo THASG-VY 98-064 Pago to Hoffman 4/27/1998		
ON-3164	Delete	1
OP-2124		43
VYC-0018	Delete	
VYC-0019		
VYC-0417		0,1
VYC-0717B		
VYC-0808	Delete	2
VYC-1254		2
VYC-1290	Delete	0,2
VYC-1388		1,0
VYC-1389		0
VYC-1442		
VYC-1628		0
VYC-1685	Delete	
VYC-1751		
VYS 98/58		
VYS 98/97		

Output Documents:

CORE SPRAY DESIGN
BASIS DOC
DBD CS
DBD RHR
RESIDUAL HEAT
[illegible]MOVAL DESIGN
[illegible]IS
VYC-0019
VYC-0417 Delete
VYC-1628
VYC-1670
VYC-1678 Delete
VYC-1685
VYC-1751
VYC-1803
VYPC 96-015

Related Programs:

Comments

Outstanding VYDEP 022 memo VYS 97/60 Delete (incorporated in Revision 6)

7.7

TABLE OF CONTENTS

1.0 OBJECTIVE

1.1 To determine the NPSH margin for the Core Spray and Residual Heat Removal (RHR) pump taking suction from the Torus (1) at maximum run out flow during the first 10 minutes following a LOCA, (2) at design flow at the maximum peak post-LOCA suppression pool temperature, and (3) over the long-term post-accident suppression pool heatup and cooldown transient.

Revision 6 differs from Revision 5 as follows:

a) CCN 1 incorporated
b) CCN 2 incorporated
c) CCN 3 incorporated
d) Assesses NPSH margins under minimum flow conditions.
e) Incorporates results of VYC-1919 (Reference 6) and VYC-1924 (Reference 2).

The RHR and Core Spray Systems are Safety Class 2.

2.0 METHOD OF SOLUTION

2.1 Required NPSH can be obtained from the pump curves based on witnessed tests performed by the pump vendor. There is a separate set of test data for each of the two Core Spray pumps and the four RHR pumps delivered to Vermont Yankee. A curve fit bounding the required NPSH data was developed for each pump type. The required NPSH for the RHR pumps is based on the data labeled "Minimum Operable NPSH @ Reduced Head". (The basis for the RHR required NPSH was reviewed during the AE Inspection (Reference 4)).

At Vermont Yankee's request, the pump vendor performed additional NPSH evaluations for the RHR and Core Spray pumps in order to provide a more rigorous basis for interpolating between and extrapolating beyond the NPSH data base provided with the pumps. The pump vendor supplemented the data supplied for each pump with additional and more extensive data for the same or similar pumps obtained from their archives. The pump vendor used these data, adjusted as necessary using pump affinity laws, to develop characteristic NPSH curves that complement the original witnessed test data. When pumps have been NPSH tested over a small flow range and NPSH data are required outside of this range, the NPSH curves have to be extrapolated. Only NPSH tests of pumps of similar style, design, specific speed, suction specific speed, number of impeller vanes and suction vane angles can be used for this purpose.

The pump vendor also provided additional information to define allowable times of operation and minimum allowable NPSH for minimum flow conditions and at higher flow rates. The vendor report is included as Attachment 5.

2.1.1 RHR Pumps

The original witnessed test data for the four (4) RHR pumps covered a flow range of 6,300 gpm to slightly less than 9,000 gpm. This range is adequate for the RHR pumps since the expected maximum flow rates following a LOCA are between 7,100 gpm and 7,400 gpm.

NPSH tests were also performed on one of the four RHR pumps, prior to the final impeller trimming, at 6300, 8065, and 9502 gpm. Five (5) to eight (8) test points were taken at each of the above capacities to establish the slope and shape of NPSH vs. Total Dynamic Head (TDH) curve. These tests established that the so-called knee of the NPSH vs. TDH curve is gradual, i.e. there is no rapid drop in TDH for a relatively small reduction in NPSH. Data points were taken at TDH reductions of up to 8% relative to the values obtained at higher values of available NPSH. These data were used to develop a family of curves of NPSH vs. flow for TDH drops of 1%, 3%, and 6%. The witnessed test data compared to these curves fell somewhere between 3% and 6% lines, and slightly below the 6% line for flow rates above 7,000 gpm for the data points labeled "Minimum Operable NPSH @ Reduced Head". The vendor concluded that the pumps, if operated with the minimum NPSH, are within acceptable limits of the NPSH knee.

Extrapolation of required NPSH to flow rates less than 6,300 gpm was based on data from tests on similar pump designs, but of different sizes (18x24x28 CVIC and 8x10x21 CVIC vs. 16x18x26 CVIC). Similar pumps are of the same suction specific speed, number of vanes and suction vane angle. The extrapolation is based on estimation and experience from NPSH tests on other styles of pumps performed in recent years, when more detailed NPSH tests were required. NPSH at lower flow rates is of less importance than at higher flow rates since available NPSH will always be higher at lower flow rates because of lower head loss due to flow, and also since core and containment cooling requirements dictate flow rates higher than 6,300 gpm when reactor water level is below the elevation of the top of the active fuel. The extrapolated NPSH at flow rates less than 6,300 gpm is not used for the evaluation of design NPSH margins. The required NPSH at low flow rates is mainly of interest in evaluating operating characteristics under minimum flow conditions, when the pumps are operating with the minimum flow bypass valves in their open position until reactor pressure drops low enough to allow the injection valves to open.

The pump vendor also provided an assessment of the potential for permanent pump damage due to cavitation at the minimum NPSH. Their assessment required input from Vermont Yankee on the durations of operation at minimum NPSH conditions. Vermont Yankee provided this information in the form of predicted suppression pool temperature and available NPSH vs. time for the LOCA. Relative to the RHR pumps, when operating for seven (7) hours at 7,000 gpm with an available NPSH of 23 to 24 feet, the pump vendor concluded that, "Depending on water temperature and

water chemistry there can be some "frosting (e.g. light pitting) on the impeller suction vanes, but there will be no detrimental pump damage due to cavitation when operating at minimum NPSH for the specified hours of operation." The vendor extended this assessment beyond the seven (7) hours at 23 to 24 feet of NPSH to define the NPSH required based on an impeller life of 8,000 hours. This information is compared to the expected decrease in suppression pool temperature following the peak, and subsequent increase in available NPSH, and it shows that the RHR pump will always operate within the acceptable bounds defined by the vendor.

The vendor recommended minimum flow requirements were given as ≤ 4 hours at 350 gpm and ≥ 4 hours at 2700 gpm. The corresponding required NPSH values were 30 ft at 350 gpm and 26 ft at 2700 gpm (page 5, Attachment 5).

2.1.2 CS Pumps

The original witnessed test data for the CS pumps covered a flow range of 3,000 gpm to slightly more than 3,800 gpm. The expected maximum flow rates following a LOCA are between 3,000 gpm and 4,600 gpm.

More comprehensive NPSH tests were performed on an identical pump for a different customer. These tests were performed at approximately 1780 rpm. Converted to 3582 rpm using affinity laws, the flow rates were 3005, 4037, 5038, 5120, 6000, 6020, and 6524 gpm, thus bounding the flow range of interest. Four (4) to ten (10) test points at each of the above capacities established the slope and shape of the NPSH vs. TDH characteristic curve. Differences in impeller trim diameter were also factored into the developed required NPSH curves. The vendor concluded that these tests were sufficient to develop NPSH characteristics for the pump and are representative of the pumps delivered to Vermont Yankee. The vendor also concluded that Vermont Yankee's pumps, if operated with the minimum NPSH, are within acceptable limits of the NPSH knee.

Extrapolation of required NPSH to flow rates less than 3,000 gpm was based on data from tests on similar pump design, but of different size (12x14x14 ½ CVDS and 14x16x23 CVDS vs. 12x16x14 CVDS). Similar pumps are of the same suction specific speed, number of vanes and suction vane angle. The extrapolation is based on estimation and experience from NPSH tests on other styles of pumps performed in recent years, when more detailed NPSH tests were required. NPSH at lower flow rates is of less importance than at higher flow rates since available NPSH will always be higher at lower flow rates because of lower head loss due to flow, and also since core cooling requirements dictate flow rates higher than 3,000 gpm when reactor water level is below the elevation of the top of the active fuel. The extrapolated NPSH at flow rates less than 3,000 gpm is not used for the evaluation of design NPSH margins. The required NPSH at low flow rates is mainly of interest in evaluating operating characteristics under minimum flow conditions, when the pumps are operating with the minimum flow bypass valves in their open position until reactor pressure drops low enough to allow the injection valves to open.

The pump vendor also provided an assessment of the potential for permanent pump damage due to cavitation at the minimum NPSH. Their assessment required input from Vermont Yankee on the durations of operation at minimum NPSH conditions. Vermont Yankee provided this information in the form of predicted suppression pool temperature and available NPSH vs. time for the LOCA. The vendor concluded that the CS pumps have more margin than the RHR pumps relative to potential damage from cavitation at the minimum available NPSH predicted for the LOCA. The vendor developed a curve of allowable hours of operation vs. available NPSH for the CS pump similar to that developed for the RHR pump. Comparison of predicted minimum available NPSH for the CS pumps vs. time for the LOCA shows that the CS pump will always operate within the acceptable bounds defined by the vendor based on an 8,000 hour impeller life.

The vendor recommended minimum flow requirements were given as ≤ 4 hours at 300 gpm and ≥ 4 hours at 1250 gpm. The corresponding required NPSH values were 32.5 ft at 300 gpm and 27 ft at 1250 gpm (page 5, Attachment 5).

2.2 Since the vendor chose to use a postulated minimum available NPSH rather than the minimum tested NPSH for their evaluation of acceptable durations of operation based on potential cavitation damage to the pump impeller, a reassessment of NPSH margins using minimum available NPSH rather than minimum tested NPSH was performed. The following table compares the values used in the vendors evaluations and comparable values at the same flow rates from the witness tests.

PUMP	FLOW (gpm)	Minimum Available NPSH assumed by Vendor for the Impeller Life Study (ft)	Minimum Required NPSH from Original Witness Tests[1] (ft)
CS	3,000	24.0	24.0
	4,600	28.0	No Data
RHR	6,400	23.0	23.2
	7,000	23.5	22.6
	7,600	24.0	23.5

Using the minimum available NPSH values from the impeller life study as required values is conservative since the values are based on the long-term reliability of the pump impellers, and they are equal to or bound the witnessed test data.

NPSH values at other flow rates are based on curve fits developed from the above data points and vendor predicted characteristic curves (NPSH vs. flow rate).

2.2.1 For CS, the curve fit incorporates the witnessed test data points for flow rates between 3,000 gpm and 4,600 gpm.

FLOW (gpm)	NPSH (ft)	SOURCE
3003	24.0	Curve No. 27692
3522	24.9	Curve No. 27692
3810	25.0	Curve No. 27692
3000	24.0	Curve No. 27691
3542	24.5	Curve No. 27691
3810	25.0	Curve No. 27691
4600	28.0	SBPI Document No. E12.5.561 (Attachment 5)

Curve Fit:

A following second order polynomial curve fit to conservatively bound the above data was developed:

[1]NPSH is determined from a curve fit that bounds all data points for each pump.

$$NPSH = 26.4 - 2.965 \times 10^{-3} Q + 7.191 \times 10^{-7} Q^2$$

where Q is Flow Rate in gpm.

Comparison between Data and Curve Fit

FLOW	DATA	FIT
3003	24.0	24.0
3500	none	24.8
3522	24.9	24.9
3810	25.0	25.5
4600	28.0	28.0

2.2.2 RHR

A different approach is taken for RHR. The witness test data were not used to develop a curve fit. The SBPI recommended minimum available NPSH bounded the Minimum Operable NPSH @ Reduced Flow data from the witness tests over the flow range of interest, therefore the recommended minimum available NPSH was used exclusively. A simple linear interpolation scheme is used as the curve fit.

Flow range: 6,400 gpm to 7,600 gpm
Sources: SBPI Document No. E12.5.561, NPSH/Minimum Flow Study-Summary Report, dated May 26, 1998. (Attachment 5)

Data:

FLOW	NPSH
6400	23.0
7000	23.5
7600	24.0

Curve Fit:

$$NPSH = 23.0 + (Q - 6400) / 1200$$

2.3 Available NPSH is calculated using the industry standard equation (Reference 1)

$$\text{NPSH Available} = (p_{Torus} - p_v)(144)v + Z - H_f - H_d - H_s$$

where p_{Torus} = Torus pressure, psia
p_v = vapor pressure of the pumped fluid, psia

v = specific volume of the pumped fluid, cu ft / lb
Z = elevation head, torus to pump suction, ft
H_s = suction strainer loss, ft
H_d = strainer debris loss, ft
H_f = friction loss in suction piping, ft

2.4 The amount of debris on a strainer will vary with time, starting at zero and increasing to the total value as the debris passes through each strainer and is removed from the water. Assuming the debris is uniformly dispersed in the suppression pool, the fraction of debris deposited on the strainers at a time, t, after initiation of flow, assuming a constant flow rate, is (see Attachment 1 for the derivation)

$$D/D_{total} = (1 - e^{-Qt/V})$$

where D/D_{total} = fraction of the total debris deposited on the strainer
Q = total pump flow rate
t = time
V = suppression pool volume = 68,000 cu ft minimum (TS 3.7.A.1.e)

This equation is used to determine the amount of fibrous debris deposited on the strainers during the first 10 minutes following a LOCA.

After ten minutes, it is assumed that one Core Spray pump provides cooling to the core, and one RHR pump cools the suppression pool. The remaining debris in the suppression pool and any debris deposited on an active strainer supplying pump(s) in the short-term that is subsequently secured for the long-term is deposited on the two active strainers in proportion to their flow rates. The total debris thus deposited on the two active strainers is used to determine NPSH margin at the peak suppression pool temperature.

2.5 A survey of ECCS single failures was done to identify what single failure resulted in the maximum debris accumulation on the strainers during the first ten minutes and at the peak suppression pool temperature.

3.0 INPUTS AND ASSUMPTIONS

3.1 The suppression pool temperatures used in the analysis are based on Reference 19

3.2 The head loss term for the RHR and Core Spray strainers is based on the DBA LOCA Base Case documented in Reference 2.

RHR @ 7400 gpm	0.33 ft	RHR @ 14200 gpm	0.48 ft
CS @ 3500 gpm	0.21 ft	CS @ 4600 gpm	0.32 ft

These head loss values are based on debris loads that are different than those calculated in VYC-1677 (Reference 3), and on a peak suppression pool temperature that is less than calculated in Reference 19. Both of these effects cause a slightly higher calculated head loss, and thus represent additional conservatism in the calculation. The degree of conservatism is indicated in the "Revised ECCS Suction Strainer Head Loss Assessment for Vermont Yankee" (Attachment 2). The calculation of NPSH margin is done at the peak LOCA suppression pool temperatures because the negative effect of vapor pressure on NPSH margin significantly outweighs the slight benefit on the debris head loss term.

3.3 The calculation also conservatively assumes that containment pressure is equal to 14.7 psia regardless of the temperature and the initial pressure. This assumption is in accordance with Regulatory Guide 1.1.

3.4 The calculation is done for two conditions called short-term and long-term.

3.4.1 The short-term condition assumes that the suppression pool temperature is at its highest calculated value at 10 minutes, and there has been no operator action to initiate suppression pool cooling or to secure or throttle ECCS pumps. Reactor pressure is assumed to be equal to containment pressure, thus ECCS pumps are operating at their maximum flow rates. The maximum suppression pool temperature at the end of 10 minutes is assumed to be 164°F (page 2, Ref. 19). The debris loading on the ECCS pump suction strainers is based on the maximum fraction of the suppression pool volume that has passed through the strainers during the first 10 minutes of the event. The maximum run out flow for the one RHR pump is 7,400 gpm in one loop, for two RHR pumps in one loop, 14,200 gpm, and for the Core Spray pump, 4,600 gpm (Ref. 7).

3.4.2 The long-term condition is assumed to represent the conditions when the peak suppression pool temperature is reached, several hours after initiation of the LOCA. The peak suppression pool temperature is assumed to be 182.6°F (Table 2, Ref. 19). The peak temperatures were selected to bound both short- and long-term conditions. It is assumed that the ECCS suction strainers have reached their maximum debris loadings by the time the suppression pool reaches its peak temperature. It is also assumed that, in accordance with operating procedures (Ref. 8), operators have initiated torus cooling with the RHR. Previous calculations of NPSH for the RHR assumed that the RHR flow would be throttled to 7,000 gpm, per procedures. In anticipation of a potential change which will eliminate the 7,000 gpm limit on flow, the present calculation will assume an RHR flow rate of 7,400 gpm, which is the maximum short-term flow rate for one RHR pump in the LPCI mode (see Section 3.3). The actual flow rate is expected to be less than 7,400 gpm.

The Core Spray pump is assumed to be throttled based on Emergency Operating Procedures. Operators will monitor NPSH limit curves in the EOPs and will throttle flow if indicated flow and pool temperature are outside the acceptable operating envelope. In order to assure adequate core

cooling, a minimum indicated flow rate of 3,244 gpm will be maintained (Ref. 20). A minimum indicated flow rate of 3,244 gpm could result in a maximum actual flow rate of 3,500 gpm, allowing about 100 gpm for an operator tolerance band and worst case flow instrument uncertainty (Table 10, Ref. 21).

3.5 The elevation head, Z, is based on the calculated torus volume at 10 minutes and at the peak pool temperature from VYC-1628 (Run 3, Ref. 23), and the relationship between volume and level from VYC-1254, Rev. 3 (Ref. 24). The pool volumes from Ref. 23 do not include the volume of water in the downcomers. Therefore, the appropriate relationship between level and volume in Ref. 24 is from Table 4.3-1 using the volumes from the column labeled "downcomers empty". The constants are the elevation of the suction datum for the Core Spray and RHR pumps.

The elevation head, Z, is the difference between the elevation of the suppression pool surface and the pump suction. Key dimensions from Dwg. 6202-1 (Ref. 13):

Torus Centerline Elevation	230 ft 1.5 in
Torus I.D.	27 ft 8 in

Therefore, elevation of torus invert = 230' 1.5" - 1/2 (27' 8") = 216' 3.5", or 216.29 ft.

	Pool Volume	Torus Water Level	Torus Water Elevation
Short-term	76,800 ft^3	11.93 ft	228.22 ft
Long-term	77,640 ft^3	12.03 ft	228.32 ft

Core Spray pump suction center line	215' 9" (215.75 ft)	Ref. 25
RHR pump suction center line	215' 11" (215.92 ft)	Ref. 26

Therefore, Z for Core Spray and RHR,

	Core Spray	RHR
Short-term	12.47 ft	12.30 ft
Long-term	12.57 ft	12.40 ft

3.6 The strainer head loss is based on the vendor calculated values for the clean strainer and fittings (Reference 6).

RHR	Core Spray
0.33 ft @ 7400 gpm	0.38 ft @ 4000 gpm
1.22 ft @ 14200 gpm	0.51 ft @ 4600 gpm

At flow rates less than 4000 gpm for Core Spray, the head loss is adjusted by the square of the ratio of the flow rate to the reference value, thus $H_s = 0.38\ (Q/4000)^2$.

3.7 The head loss in the suction piping from the torus to the pumps is based on the calculations in Attachment 6 for Core Spray and Attachment 7 for RHR. These calculations are adjusted here to remove the terms for the old strainer tees and other fittings associated with the strainers inside the torus since the guaranteed head loss term for the new strainers includes those fittings already, therefore the adjusted values will represent only the piping runs and fittings from the new strainer to the pumps.

Friction and form losses in suction piping, including the old strainer tee entrance, are from Attachment 6 for Core Spray. From p. 3 of Attachment 6, the L/D for the "strainer entrance tee" was given as 30. The total L/D for all fittings was 132. Deducting the strainer entrance tee leaves a revised total L/D of 102. The total equivalent length of 12" (STD) pipe is 102 ft plus 35 ft for the pipe run (from p. 2 of Attachment 6), or 137 ft. This length is then adjusted to 12" Schedule 40 pipe by multiplying by 0.974 (p. 4, Attachment 6), thus 137 ft x 0.974 = 133 ft. of 12" Schedule 40 equivalent. The head loss for 3000 gpm is, from p. 4, Attachment 6,

$$H_{fCS} = 133\text{ ft } (0.731\text{ psi}/100\text{ ft})\ (2.31\text{ ft/psi}) = 2.25\text{ ft @ }3000\text{ gpm}$$

or, generalizing,

$$H_{fCS} = 2.25\ (Q/3000)^2 = 2.5 \times 10^{-7}\ Q_{CS}^2$$

For RHR, friction and form losses in suction piping, including the old strainer tee entrance, are from Attachment 7. From p. 3 of Attachment 7, the L/D for the "strainer entrance tee" and "Miter Bend" was given as 119 and 6, respectively, in terms of 24" STD pipe. The total L/D for all fittings was 153. Deducting the strainer entrance tee and miter bend leaves a revised total L/D of 28. Converting to the equivalent length of 24" STD pipe,

$$L_{24"} = (28)(23.25"/12"/\text{ft}) = 54.25\text{ ft}$$

This value is added to the 8.73 ft of piping run from p. 2, Attachment 7, for a total length of 62.98 ft. Attachment 7 next adjusted pipe lengths to an equivalent length of 20" Schedule 40 pipe using an equivalent pressure drop basis. The conversion factor is 0.347 from p. 3 of Attachment 7, thus

$$L_{20"Sch40} = 0.347\ (62.98\ \text{ft}) = 21.85\ \text{ft}.$$

This value is next added to the equivalent length of 26" pipe in terms of 20"Schedule 40, which is 39.90 ft (p.5, Attachment 7). Thus, the total equivalent length of 20"Schedule 40 pipe from the torus to the tee connection to the RHR pumps is 21.85 + 39.90 = 61.75 ft, excluding the old strainer tee and miter bend. This value can be carried through the remainder of the calculations in Attachment 7, and arrive at the following expressions for single pump and two pump operation.

For single pump operation, refer to p. 11, Attachment 7,

$$H_{f,1RHR} = 4.77 \times 10^{-8}\, Q^2$$

For two pump operation, refer to p. 12, Attachment 7,

$$H_{f,2RHR} = 7.836 \times 10^{-8}\, Q^2$$

here Q in both cases refers to RHR pump flow per pump in gpm.

4.0 CALCULATION

4.1 Debris Accumulation

4.1.1 The following ECCS combinations, based on single failure (including none), were considered in determining short-term debris accumulation (the use of the designations "A" and "B" is arbitrary).

Single Failure	No. of CS Pumps	No. of RHR Pumps, Loop A	No. of RHR Pumps, Loop B	No. of Active Suction Sites, N_s
Diesel Generator	1	1	1	3
CS Pump/Injection Valve	1	2	2	3
LPCI Injection Valve	2	2	0	3
RHR Pump	2	2	1	4
None	2	2	2	4

4.1.2 The run out flow rates, per pump, using the values from Paragraph 3.4,

Single Failure	CS A	CS B	RHR per pump Loop A	Total RHR, Loop A	RHR per pump Loop B	Total RHR, Loop B
Diesel Generator	4,600	0	7,400	7,400	7,400	7,400
CS Pump/Injection Valve	4,600	0	7,100	14,200	7,100	14,200
LPCI Injection Valve	4,600	4,600	7,100	14,200	0	0
RHR Pump	4,600	4,600	7,100	14,200	7,400	7,400
None	4,600	4,600	7,100	14,200	7,100	14200

4.1.3 The total flow rate from the suppression pool to the reactor vessel, and the debris fraction at ten minutes, using the equation from Paragraph 2.4,

Single Failure	Total Flow Rate, Q	$D/D_{total} = (1 - e^{-Qt/V})$ t=10 min V=68,000 ft^3 x 7.48 gal/ft^3
Diesel Generator	4,600+7,400+7,400=19,400	0.317
CS Pump/Injection Valve	4,600+14,200+14,200=33,000	0.477
LPCI Injection Valve	9,200+14,200=23,400	0.369
RHR Pump	9,200+14,200+7,400=30,800	0.454
None	9,200+14,200+14,200=37,600	0.522

4.1.4 After 10 minutes, all but one Core Spray pump and one RHR pump are assumed to be secured. In addition, the Core Spray pump is assumed to be throttled to 3500 gpm. The distribution of debris on the one active CS strainer and the one active RHR strainer will be the amount initially deposited in the short-term, plus the amount redistributed from the now-inactive strainer(s), plus the amount not removed in the short-term. The distribution of the remaining amounts will be in proportion to the CS and RHR flow rates. The results are summarized below.

Single Failure		CS A	CS B	RHR A	RHR B	Total
Diesel Generator	Short Term Flow Rate	4600	0	7400	7400	19400
	Short Term Accumulation	0.075	0.000	0.121	0.121	0.317
	Long Term Flow Rate	3500	0	7400	0	10900
	Redistributed	0.000	0.000	0.000	0.121	0.121
	Long Term Accumulation	0.333	0.000	0.667	0.000	1.000
CS Pump/Inj. Valve	Short Term Flow Rate	4600	0	14200	14200	33000
	Short Term Accumulation	0.066	0.000	0.205	0.205	0.477
	Long Term Flow Rate	3500	0	7400	0	10900
	Redistributed	0.000	0.000	0.000	0.205	0.205
	Long Term Accumulation	0.300	0.000	0.700	0.000	1.000
LPCI Inj. Valve	Short Term Flow Rate	4600	4600	14200	0	23400
	Short Term Accumulation	0.073	0.073	0.224	0.000	0.369
	Long Term Flow Rate	3500	0	7400	0	10900

	Redistributed	0.000	0.073	0.000	0.000	0.073
	Long Term Accumulation	0.298	0.000	0.702	0.000	1.000
RHR Pump	Short Term Flow Rate	4600	4600	14200	7400	30800
	Short Term Accumulation	0.068	0.068	0.209	0.109	0.454
	Long Term Flow Rate	3500	0	7400	0	10900
	Redistributed	0.000	0.068	0.000	0.109	0.177
	Long Term Accumulation	0.300	0.000	0.700	0.000	1.000
None	Short Term Flow Rate	4600	4600	14200	14200	37600
	Short Term Accumulation	0.064	0.064	0.197	0.197	0.522
	Long Term Flow Rate	3500	0	7400	0	10900
	Redistributed	0.000	0.064	0.000	0.197	0.261
	Long Term Accumulation	0.301	0.000	0.699	0.000	1.000

The above information was used to determine the distribution of each of the various debris species postulated to be deposited in the suppression pool following a LOCA (Reference 3). However, the above distributions were not used to determine the head loss due to debris used in this calculation as discussed below. The above distributions were used as input to VYC-1677 (Reference 3), and the resulting debris loads were used to assess the head loss due to debris shown in Attachment 2.

4.2 Head Loss due to Debris

The maximum predicted head loss for the CS and RHR strainers are based on the vendor calculations (Reference 2), using conservative debris loads, fluid temperatures, and flow rates. These were discussed in Section 3.2 as Inputs and Assumptions. The head losses so determined are shown in Attachment 2 to be conservative relative to updated information on debris distribution (Section 4.1), debris loads (Reference 3), flow rates and fluid temperatures. Attachment 2 has not been adopted as the design basis because additional assessments are ongoing regarding the time interval for torus cleaning, which may affect the final specification of the "sludge" term in the head loss calculations.

4.3 NPSH Margin

The Table 1 summarizes the calculated available NPSH at the flow rates and temperatures of interest for a clean strainer. The calculations were done for the design basis flow rates and maximum short-term and long-term temperatures. Other temperature points were calculated to show the sensitivity of the results. The available NPSH is compared to the required NPSH, and if the margin is greater than the maximum head loss due to debris, then it can be concluded that post-accident performance is acceptable at the design basis points.

Table 2 provides similar results for the minimum flow mode. Since Technical Specifications require reactor depressurization when suppression pool temperature reaches 120 F, it is unlikely that a CS or RHR pump would be operating in a minimum flow mode for very long at that temperature. Table 2 shows adequate NPSH margin for pool temperatures ≥ 164 F.

Attachment 3 provides an assessment of the time-dependent NPSH margin for the maximum post-accident pool temperature transient. The purpose of this calculation was to show the margin available after the peak temperature is reached in light of the vendor recommended increase in minimum available NPSH after 7 hours. The minimum available NPSH is shown to be above the vendor recommended minimum at all times during the most limiting transient.

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NPSHA = (14.7 - P_g)(144)(v_f) + Z - h_f - h_s - h_d

<u>Suction Line Losses</u>

1 RHR	h_f = 4.77e-8*Q^2	CS	h_f = 2.5e-7*Q^2
2 RHR	h_f = 7.84e-8*(Q/2)^2		

<u>Clean Strainer Losses</u>

1 RHR	h_s = 0.33	CS	h_s = 0.51 for Q=4600 gpm
2 RHR	h_s = 1.22	CS	h_s = 0.38*(Q/4000)^2 for Q≤4000 gpm

<u>Maximum Debris Losses</u>

1 RHR	h_d = 0.33 (maximum)	CS	h_d = 0.21 (maximum @ 4000 gpm)
2 RHR	h_d = 0.48 (maximum)	CS	h_d = 0.32 (maximum @ 4600 gpm)

<u>Required NPSH</u>

RHR	NPSHR = 23 + (Q-6400)/1200	CS	NPSHR = 26.4 - 2.965e-3*Q +7.191e-7*Q^2

	Q (gpm)	T (F)	P_g (psia)	v_f (cu ft/lb)	Z (ft)	h_f (ft)	h_s (ft)	h_d (ft)	"Clean" NPSHA (ft)	NPSHR (ft)	"Clean" Margin (ft)	h_d (ft)	Margin w/ Debris (ft)
1 CS	4600	90	0.69813	0.016099	12.49	5.29	0.51	0	39.1	28.0	11.2		
short-term	4600	120	1.6927	0.016204	12.49	5.29	0.51	0	37.0	28.0	9.1		
	4600	150	3.7184	0.016343	12.49	5.29	0.51	0	32.5	28.0	4.6		
	4600	164	5.212	0.016410	12.49	5.29	0.51	0	29.1	28.0	1.1	0.32	0.8
	4600	182.6	7.954	0.016530	12.49	5.29	0.51	0	22.7	28.0	-5.2		
	4600	190	9.34	0.016572	12.49	5.29	0.51	0	19.5	28.0	-8.5		
1 CS	3500	90	0.69813	0.016099	12.59	3.06	0.29	0	41.7	24.8	16.9		
long-term	3500	120	1.6927	0.016204	12.59	3.06	0.29	0	39.6	24.8	14.8		
	3500	150	3.7184	0.016343	12.59	3.06	0.29	0	35.1	24.8	10.2		
	3500	164	5.212	0.016410	12.59	3.06	0.29	0	31.7	24.8	6.8		
	3500	182.6	7.954	0.016530	12.59	3.06	0.29	0	25.3	24.8	0.5	0.21	0.3
	3500	184	8.203	0.016535	12.59	3.06	0.29	0	24.7	24.8	-0.1		
	3500	185	8.384	0.016541	12.59	3.06	0.29	0	24.3	24.8	-0.6		
2 RHR	14200	90	0.69813	0.016099	12.32	3.95	1.22	0	39.6	23.6	16.0		
short-term	14200	120	1.6927	0.016204	12.32	3.95	1.22	0	37.5	23.6	13.9		
	14200	150	3.7184	0.016343	12.32	3.95	1.22	0	33.0	23.6	9.4		
	14200	164	5.212	0.016410	12.32	3.95	1.22	0	29.6	23.6	6.0	0.48	5.5
	14200	182.6	7.954	0.016530	12.32	3.95	1.22	0	23.2	23.6	-0.4		
	14200	190	9.34	0.016572	12.32	3.95	1.22	0	19.9	23.6	-3.6		
1 RHR	7400	90	0.69813	0.016099	12.32	2.61	0.33	0	41.8	23.8	18.0		
short-term	7400	120	1.6927	0.016204	12.32	2.61	0.33	0	39.7	23.8	15.9		
	7400	150	3.7184	0.016343	12.32	2.61	0.33	0	35.2	23.8	11.4		
	7400	164	5.212	0.016410	12.32	2.61	0.33	0	31.8	23.8	8.0	0.33	7.6
	7400	182.6	7.954	0.016530	12.32	2.61	0.33	0	25.4	23.8	1.6		
	7400	190	9.34	0.016572	12.32	2.61	0.33	0	22.2	23.8	-1.7		
1 RHR	7400	90	0.69813	0.016099	12.42	2.61	0.33	0	41.9	23.8	18.1		
long-term	7400	120	1.6927	0.016204	12.42	2.61	0.33	0	39.8	23.8	16.0		
	7400	150	3.7184	0.016343	12.42	2.61	0.33	0	35.3	23.8	11.5		
	7400	164	5.212	0.016410	12.42	2.61	0.33	0	31.9	23.8	8.1		
	7400	182.6	7.954	0.016530	12.42	2.61	0.33	0	25.6	23.8	1.7	0.33	1.4
	7400	185	8.384	0.016541	12.42	2.61	0.33	0	24.5	23.8	0.7		
	7400	190	9.34	0.016572	12.42	2.61	0.33	0	22.3	23.8	-1.6		

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TABLE 2

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$NPSHA = (14.7 - P_g)(144)(v_f) + Z - h_f - h_s - h_d$

<u>Suspend</u>

Suction Line Losses

1 RHR	$h_f = 4.77e\text{-}8 \cdot Q^2$	CS	$h_f = 2.5e\text{-}7 \cdot Q^2$
2 RHR	$h_f = 7.84e\text{-}8 \cdot (Q/2)^2$		

Clean Strainer Losses

1 RHR	$h_s = 0.33 \cdot (Q/7400)^2$	CS	$h_s = 0.38 \cdot (Q/4000)^2$
2 RHR	$h_s = 1.22 \cdot (Q/14200)^2$		

Required NPSH

RHR	NPSHR =	30	@ 350 gpm	CS	NPSHR =	32.5	@ 300 gpm
RHR	NPSHR =	26	@ 2700 gpm	CS	NPSHR =	27	@ 1250 gpm

	Q (gpm)	T (F)	P_g (psia)	v_f (cu ft/lb)	Z (ft)	h_f (ft)	h_s (ft)	NPSHA (ft)	NPSHR (ft)	"Clean" Margin (ft)
1 CS	300	90	0.69813	0.016099	12.47	0.0225	0.00	44.9	32.5	12.4
min-flow	300	120	1.6927	0.016204	12.47	0.0225	0.00	42.8	32.5	10.3
≤ 4 hr	300	150	3.7184	0.016343	12.47	0.0225	0.00	38.3	32.5	5.8
	300	164	5.212	0.016410	12.47	0.0225	0.00	34.9	32.5	2.4
	300	181.9	7.836	0.016521	12.47	0.0225	0.00	28.8	32.5	-3.7
	300	190	9.34	0.016572	12.47	0.0225	0.00	25.2	32.5	-7.3
1 CS	1250	90	0.69813	0.016099	12.47	0.39	0.04	44.5	27.0	17.5
min-flow	1250	120	1.6927	0.016204	12.47	0.39	0.04	42.4	27.0	15.4
> 4 hr	1250	150	3.7184	0.016343	12.47	0.39	0.04	37.9	27.0	10.9
	1250	164	5.212	0.016410	12.47	0.39	0.04	34.5	27.0	7.5
	1250	182.6	7.954	0.016530	12.47	0.39	0.04	28.1	27.0	1.1
	1250	184	8.203	0.016535	12.47	0.39	0.04	27.5	27.0	0.5
	1250	185	8.384	0.016541	12.47	0.39	0.04	27.1	27.0	0.1
2 RHR	700	90	0.69813	0.016099	12.30	0.01	0.00	44.7	30.0	14.7
min-flow	700	120	1.6927	0.016204	12.30	0.01	0.00	42.6	30.0	12.6
≤ 4 hr	700	150	3.7184	0.016343	12.30	0.01	0.00	38.1	30.0	8.1
	700	164	5.212	0.016410	12.30	0.01	0.00	34.7	30.0	4.7
	700	181.9	7.836	0.016521	12.30	0.01	0.00	28.6	30.0	-1.4
	700	190	9.34	0.016572	12.30	0.01	0.00	25.1	30.0	-4.9
2 RHR	5400	90	0.69813	0.016099	12.30	0.57	0.18	44.0	26.0	18.0
min-flow	5400	120	1.6927	0.016204	12.30	0.57	0.18	41.9	26.0	15.9
> 4 hr	5400	150	3.7184	0.016343	12.30	0.57	0.18	37.4	26.0	11.4
	5400	164	5.212	0.016410	12.30	0.57	0.18	34.0	26.0	8.0
	5400	182.6	7.954	0.016530	12.30	0.57	0.18	27.6	26.0	1.6
	5400	190	9.34	0.016572	12.30	0.57	0.18	24.3	26.0	-1.7

5.0 SUMMARY & CONCLUSIONS

Available NPSH exceeds required NPSH for the limiting conditions evaluated, short-term and long-term. This includes margin to accommodate the maximum head loss across the Core Spray and RHR suction strainers due to the predicted accumulation of fibrous and other debris following a LOCA.

There is no need to do a Safety Evaluation since the results continue to show adequate NPSH margins for the Core Spray and RHR pumps.

The completion of this calculation also satisfies the commitment made in Reference 16 to revise the NPSH calculation using a corrected curve fit for required NPSH for the RHR pump.

Impact on Other Design Output Documents

Design Basis Documents: Both the Core Spray and RHR DBDs refer to VYC-808. A Pending Change notification for both documents has been issued. A Pending Change notification to delete VYC-1678 from the references in the RHR DBD was also issued.

VYC-0019, Rev. 1: Refers to VYC-808 for the Core Spray suction strainer head loss of 0.42 ft at 4000 gpm. This revision to VYC-808 reduces that value to 0.38 ft, but the impact on VYC-0019, Rev. 1, is minimal and no change is required.

VYC-1628, Rev. 0: Refers to VYC-808 in regards to the fact that Vermont Yankee does not credit any wetwell pressure above atmospheric in the calculation of available NPSH. This revision to VYC-808 does not change that fact.

VYC-1670, Rev. 0: Refers to VYC-808 in support of use of 5 ft as a conservative value for RHR suction strainer head loss. This revision to VYC-808 does not change that conclusion.

VYC-1677, Rev. 0: Refers to VYC-808 for the debris distribution based on the short-term and long-term flow splits documented in Section 4.1.4. This revision of VYC-808 is the first use of this information.

VYC-1803, Rev. 1: Refers to VYC-808 as the basis for the calculation of available NPSH for RHR pumps at elevated suppression pool temperatures. This revision of VYC-808 does not change that statement.

VYPC 96-015, Rev. 2: Refers to VYC-808 as the basis for concluding that there is no need to throttle an RHR pump while operating in the torus cooling mode. This revision to VYC-808 does not change that conclusion.

6.0 REFERENCES

1. American National Standard for Centrifugal Pumps, ANSI/HI 1.1-1.5-1994.

2. VYC-1924, Rev. 0, "DE&S Calc. DC-A34600.006, 'Vermont Yankee ECCS Suction Strainer Head Loss Performance Assessment, RHR and CS Debris Head Loss Calculations.'"

3. VYC-1677, "Debris Source Terms For Sizing of Replacement Residual Heat Removal and Core Spray Strainers."

4. Letter, USNRC to VYNPC, Vermont Yankee Design Inspection (NRC Inspection Report 50/271/97-201), NVY 97-130, dated August 27, 1997.

5. VYC-1628F, "Limiting Torus Temperature Response, Million Second Run."

6. VYC-1919, Rev. 0, "DE&S Calc. DC-A34600-01, "RHR and CS Suction Strainer Assembly Clean Head Loss."

7. Memo, R.E.Swenson/P.A.Rainey to M.Mills, "Evaluation of Maximum Expected Flows for ECCS Strainer Replacement," VYS 98/97, dated August 28, 1997.

8. OP-2124, Rev. 43, "Residual Heat Removal System".

9. not used

10. not used

11. not used

12. not used

13. Drawing 6202-1, Rev. 1, "General Plan--Pressure Suppression Containment Vessel."

14. not used

15. not used

16. Letter, VYNPC to USNRC, "Reply to Inspection Report No. 50-271/97-201," (Appendix B, IFI 97-201-04), BVY 97-138, dated 10/27/97.

17. not used

18. not used

19. DE&S Memo, C. D. Fago to J. R. Hoffman, "Torus Temperature Margin Assessment for Confirmatory Analyses," THSAG-VY-98-064, dated April 27, 1998.

20. Memo, B. C. Slifer to K. H. Bronson, "NPSH Limits for Emergency Operating Procedures," VYS 98/58, dated May 13, 1998.

21. VYC-717B, Rev. 0, "Core Spray Pump Discharge Flow, Supplemental Addition to Revision 1."

22. Letter, USNRC to VYNPC, "Summary of Meeting on March 24, 1998, regarding Activities at Vermont Yankee Nuclear Power Station (TAC No. MA0987)," NVY 98-43, dated March 31, 1998.

23. VYC-1628, Rev. 0, "Torus Temperature and Pressure Response to Large Break LOCA and MSLB Accident Scenarios."

24. VYC-1254, Rev. 3, "Containment RPV Volume Calculations."

25. Drawing G-191207, Rev. 10

26. Drawing G-191211, Rev. 16

Derivation of strainer debris accumulation equation.

Debris is added to the suppression pool, which is a constant volume, V. The amount of debris, D, is uniformly mixed in the pool, giving a mixture concentration, D/V. The water containing the debris is pumped through a filter, which is assumed to be 100% efficient in removing the debris from the water. The filtered water is returned to the pool. Assuming a flow rate, Q, through the filter, the rate of accumulation of debris on the strainer can be expressed as follows

$$\frac{dD_{strainer}}{dt} = Q\left(\frac{D}{V}\right)$$

The concentration, $\frac{D}{V}$, changes with time as debris is removed. The rate of change of concentration is determined by taking the derivative of (D/V) with respect to time, thus

$$\frac{d(D/V)}{dt} = \frac{1}{V}\frac{dD}{dt} - \frac{D}{V^2}\frac{dV}{dt}$$

Since V = constant, dV/dt = 0. Also, the rate of change of debris in the pool, dD/dt, equals the negative of the rate of accumulation on the strainer. Thus,

$$\frac{d(D/V)}{dt} = -\frac{1}{V}Q\left(\frac{D}{V}\right)$$

$$\frac{d(D/V)}{(D/V)} = -\frac{Q}{V}dt$$

Integrating

$$\int \frac{d(D/V)}{(D/V)} = \int -\frac{Q}{V}\,dt$$

$$\ln\left(\frac{D}{V}\right) = -\frac{Q}{V}t + \text{constant of integration}$$

Initial conditions,

when $t = 0$, $D = D_0$, $\therefore$ constant $= \ln\left(\frac{D_0}{V}\right)$

Substituting

$$\ln\left(\frac{D}{V}\right) = -\frac{Q}{V}t + \ln\left(\frac{D_0}{V}\right)$$

$$\ln\left(\frac{D}{V}\right) - \ln\left(\frac{D_0}{V}\right) = -\frac{Q}{V}t$$

$$\ln\frac{D/V}{D_0/V} = -\frac{Q}{V}t$$

$$\frac{D}{D_0} = e^{-\frac{Q}{V}t} \quad \text{or} \quad D = D_0\,e^{-\frac{Q}{V}t}$$

Now, the amount on the strainer equals the amount removed from the pool, which is $D_0 - D$.

Thus $D_{strainer} = D_0 - D = D_0 - D_0\,e^{-\frac{Q}{V}t}$

or

$$\boxed{\frac{D_{strainer}}{D_0} = \left(1 - e^{-\frac{Q}{V}t}\right)}$$

ITS/VY-00-001
Rev.0

Revised ECCS Suction Strainer Head Loss Assessment for Vermont Yankee

Prepared for
Vermont Yankee Nuclear Power Corporation

Innovative Technology Solutions Corporation
6000 Uptown Blvd. NE, Suite 300
Albuquerque, NM 87110

October 15, 2000

Prepared By:	Peter K. Mast	Signature	10/15/00 Date
Reviewed By:	Gilbert L. Zigler	Signature	10/16/00 Date

An assessment of ECCS suction strainer head loss for the Vermont Yankee plant was completed in August of 1998 (Calculation DC-A34600.006). Since that time, better information has been obtained on a couple of key parameters in the analysis. These changes are as follows:

- The long-term CS flow rate was determined to be 3500 GPM (previously, a parametric range of 4000-4600 GPM had been considered).
- The long-term pool temperature has been determined to be 185 Deg F.
- The quantity of sludge used for the analysis was reduced by approximately a factor of two.
- Minor changes in fibrous debris distribution.

As a result of the first and fourth of these changes, the distribution of debris on the RHR and CS strainers is changed; a greater fraction of the debris now is deposited on the RHR strainer. Thus, this change would be expected to result in a higher RHR strainer head loss (same flow rate, same temperature, and greater debris quantity) and a lower CS strainer head loss (lower flow rate, same temperature, and lower debris quantity).

The second of these changes, the increased water temperature, and the third of these changes, the reduction in sludge quantity, would tend to reduce head loss for both the CS and RHR strainers, assuming the same assumptions on sludge behavior (filtration) are used in the analysis.

To evaluate the impact of the above changes on strainer head loss, Cases 1 and 3b (the long-term 1-pump RHR and CS analyses, respectively) were reanalyzed. The following table summarizes the debris quantities used in this analysis along with a comparison to those previously used.

Parameter	Units	RHR System		CS System	
		Old Case 1	New Case 1	Old Case 3b	New Case 3b
Flow Rate	GPM	7400	7400	4000	3500
Water Temperature	Deg F	173	185	173	185
Nukon	Lbm	258	256	152	122
Fibermat	Cu-Ft	9.6	10	5.7	5
TempMat	Lbm	20.5	31	12.1	15
Armaflex	Cu-Ft	0 (1)	0 (1)	0 (1)	0 (1)
Sludge (dry)	Lbm	546	271	322	129
Rust	Lbm	35.3	35	20.8	17
Qualified Coat	Lbm	61	60	36	29
Unqual Coat – IOZ	Lbm	70	70	42	42
Unqual Coat – Epox	Sq-Ft	0 (2)	0 (2)	0 (2)	0 (2)

(1) – This material was shown to float and not impact strainer performance.
(2) – ARL testing showed that this coatings debris did not deposit on strainer.

Using the above distributions of fibrous insulation debris, the total mass of fibrous debris changed from 336 lbm to 347 lbm for Case 1 (RHR) and from 198 lbm to 167 lbm in Case 3b (CS). Aside from the change in the particulate debris quantities specified above, and the change in flow rate for Case 3b, all other analysis parameters used in this reassessment are identical to those used in the previous calculations. This includes a reduction of 50% in the quantity of sludge, qualified coatings debris, and IOZ debris being deposited on the strainer due imperfect filtration of this material by the relatively thin fiber mat.

The results of these analyses, along with a comparison to the previously calculated results, are presented in the following table.

Calculated Parameter	Units	RHR System		CS System	
		Old Case 1	New Case 1	Old Case 3b	New Case 3b
Head Loss	Ft water	0.33	0.26	0.21	0.08
Debris Thickness	inches	1.9	2.1	2.6	2.2

9.31 VYC 1924

Table of Contents

1.0 Introduction

The objective of this CCN is to determine the available NPSH for the Residual Heat Removal (RHR) and Core Spray (CS) pumps at design flow during the most limiting torus heat-up scenario and compare the available NPSH (NPSHa) to the required NPSH (NPSHr).

2.0 Analysis

For each of the RHR and CS pumps, NPSHa as a function of time will be determined based on methods developed in VYC-808, Revision 6. NPSHa varies with time only because the fluid temperature does so during the transient. The limiting torus temperature profile from VYC-1628F (run15) is used for this analysis. The NPSHa for each pump is then plotted against the vendor's allowable NPSHa to demonstrate pump operability.

The vendor's allowable NPSHa are shown in Figures 2.1-1 (RHR Pumps) and 2.2-1 (CS Pumps). For a complete description of these curves, see VYC-808, Revision 6 and its Attachment 5. In summary, the curves show the allowable operating period for a given pump at any NPSHa. Provided the calculated NPSHa for a specific flow rate is, at all times, greater than the corresponding curve for the same flow rate, the pump is considered operable regarding NPSH.

The pump vendor is Sulzer Bingham Pumps, Inc. (SBPI). Their allowable NPSHa are provided for flow rates of 6400, 7000, and 7600 gpm for the RHR pump (VYC-808, Attachment 5). The maximum long-term flow rate of the system is 7400 gpm (VYC-808, Section 3.4). The first step in this calculation is to develop an allowable NPSHa curve at the maximum flow. This is done using the curve fit from VYC-808, Revision 6, Section 2.2.2 for times between 1 and 7 hours, and by linear interpolation between the SBPI curves for 7000 gpm and 7600 gpm beyond 7 hours. The use of a linear interpolation beyond 7 hours is conservative based on the predicted variations among 6400, 7000, and 7600 gpm as shown on the SBPI curves. The curve fit from VYC-808 is:

$$NPSH_{RHR} = 23.0 + \frac{Q-6400}{1200}\ feet \qquad \text{(Eq. 1.1)}$$

The next step is to plot the available NPSH for the limiting torus temperature scenario. The equation for NPSHa – developed in Section 4.2.2.4 of VYC-808 for the RHR pump at 7400 gpm – is:

$$NPSHa_{RHR} = (p_{torus} - p_v)(144)v + Z - H_f - H_D - H_{strainer} \qquad \text{(Eq. 1.2)}$$

where, p_{torus} = torus pressure (14.7 psia)
p_v = vapor pressure of water at temperature (psia)
144 = conversion factor
v = specific volume of water at temperature (ft^3/lbm)
Z = elevation head (12.42 ft)
H_f = friction head loss (2.61 ft)
H_D = strainer debris head loss (0 ft)
$H_{strainer}$ = clean strainer head loss (0.35 ft)

The method of solution for the Core Spray pump is the same as that for the RHR pump, but with the following exceptions. SBPI's allowable NPSHa are provided for flow rates of 3000, 3500, and 4600 gpm for the CS pump (attached). The maximum long-term flow rate of the system is 3500 gpm (VYC-808, Section 3.4). Therefore, the minimum allowable NPSH is known and needn't be derived.

The equation for NPSHa developed in Section 4.2.3.4 of VYC-808 for the CS pump at 3500 gpm is also Equation 1.2, but with the following values:

p_{torus} = torus pressure (14.7 psia)
p_v = vapor pressure of water at temperature (psia)
144 = conversion factor
v = specific volume of water at temperature (ft^3/lbm)
Z = elevation head (12.59 ft)
H_f = friction head loss (3.06 ft)
H_D = strainer debris head loss (0 ft)
$H_{strainer}$ = clean strainer head loss (0.32 ft)

3.0 Inputs/Outputs

Design inputs used in this calculation are:

- the temperature data from VYC-1628F for the limiting torus heat-up scenario (`run15`), and
- an adder to the above temperatures that accounts for event conditions that were not modeled. From Reference 19, the most limiting adder is 0.9°F.

The assumptions used in this calculation are the same as those in VYC-808, Revision 6.

4.0 Results

4.1 RHR Pump

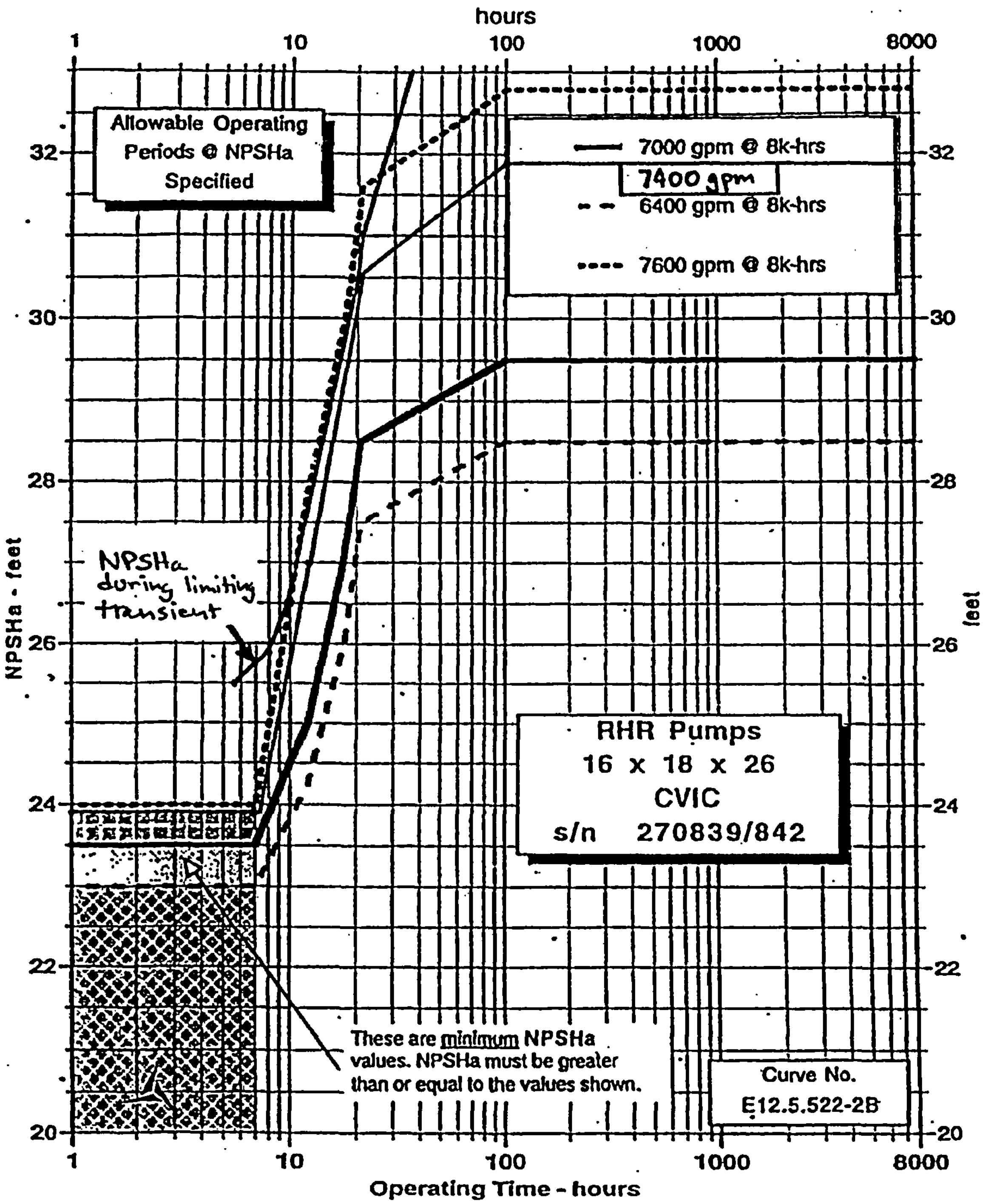
Figure 2.1-1 shows the SBPI allowable times at NPSH for RHR pump flow rates of 6400, 7000, and 7600 gpm. The same curve for the design flow rate of 7400 gpm is also shown. It is derived by using Equation 1.1 for times between 1 hour and 7 hours, and by linear interpolation between the SBPI curves for times beyond 7 hours. The data points are:

Time (hours)	NPSHa @ 7000 gpm (feet)	NPSHa @ 7600 gpm (feet)	NPSHa @ 7400 gpm (feet)
1	N/A Use Equation 1.1		23.8
7	N/A Use Equation 1.1		23.8
13.5	25.0	28.0	27.0
21	28.5	31.6	30.6
100	29.5	32.75	31.7
8000	29.5	32.75	31.7

The limiting NPSHa curve for the RHR pump is determined with Equation 1.2 and the limiting torus water temperature profile from VYC-1628F, `run15` (Reference 5). The temperatures from `run15` are increased by 0.9°F to account for event conditions that were not modeled (Reference 19). The data points for the curve are shown in Table 2.1-1. Values for p_v and v are interpolated from the saturated steam tables.

The limiting NPSHa falls above the allowable curve at the design flow rate for all times. Therefore, the pump's operability is not compromised.

Figure 2.1-1
RHR Pump Allowable Operating Periods @ NPSHa Specified



4.2 CS Pump

Figure 2.2-1 shows the SBPI allowable times at NPSH for CS pump flow rates of 3000, 3500, and 4600 gpm.

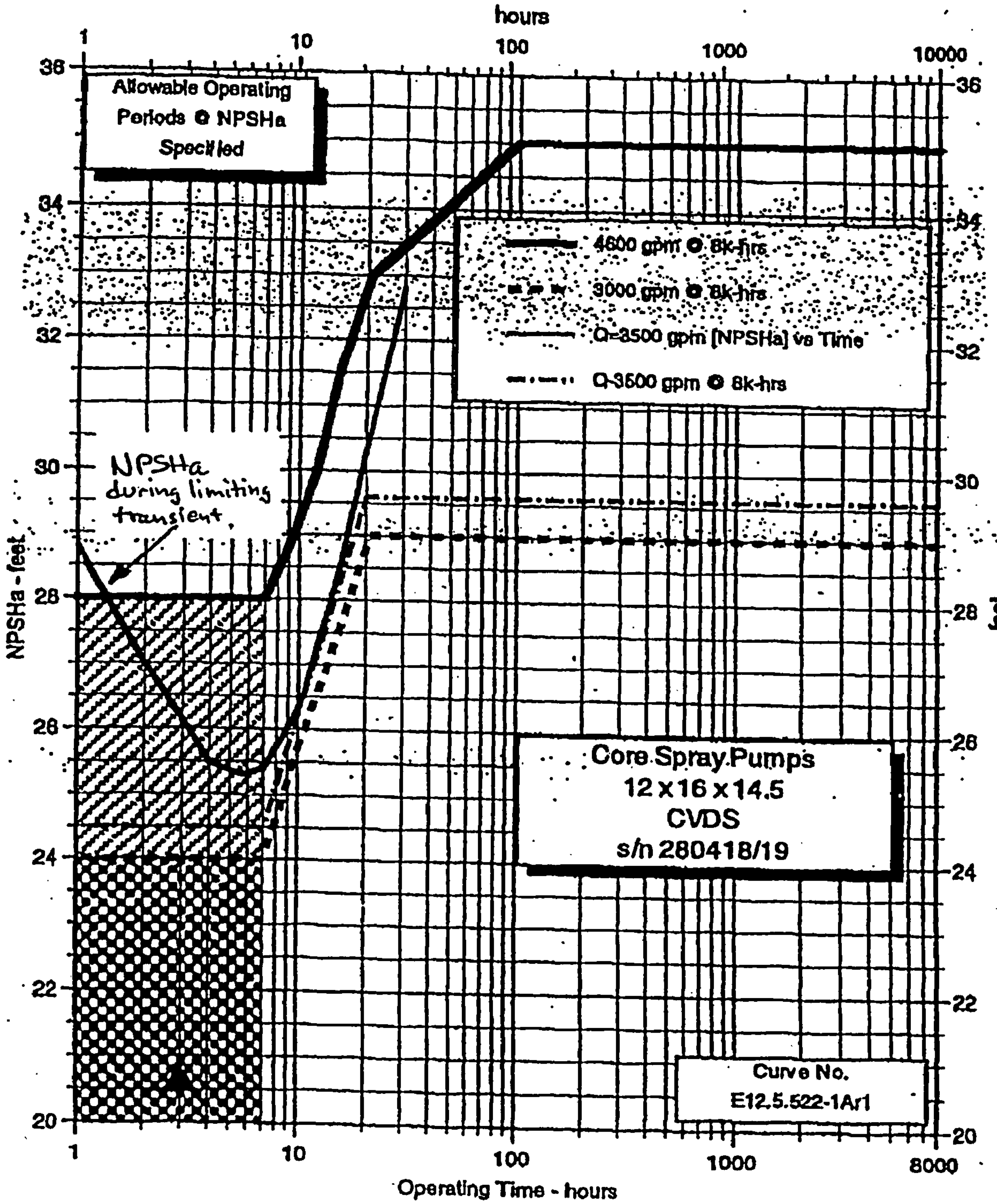
The limiting NPSHa curve for the CS pump is determined with Equation 1.2 and the limiting torus water temperature profile from VYC-1628F, run15 (Reference 5). The temperatures from run15 are increased by 0.9°F to account for event conditions that were not modeled (Reference 19). The data points for the curve are shown in Table 2.2-1. Values for p_v and v are interpolated from the saturated steam tables.

The limiting NPSHa falls above the allowable curve at the design flow rate for all times. Therefore, the pump's operability is not compromised.

Table 2.2-1
CS Pump NPSHa During the Limiting Torus Temperature Transient

Time	Temperature (°F)		p_v (psia)	v (ft³/lbm)	NPSHa (ft)
	From run15	+0.9°F			
1 hrs (3,595 sec)	172.3	173.2	6.449	0.01647	28.8
2 hrs (7,203 sec)	177.2	178.1	7.204	0.016499	27.0
4 hrs (14,410 sec)	181.0	181.9	7.836	0.016521	25.5
5.6 hrs (20,240 sec)	181.7	182.6	7.956	0.016525	25.3
7 hrs (25,194 sec)	181.4	182.3	7.904	0.016524	25.4
10 hrs (36,050 sec)	179.2	180.1	7.529	0.016511	26.3
15 hrs (54, 052 sec)	173.4	174.3	6.613	0.016476	28.4
20 hrs (72,054 sec)	167.7	168.6	5.807	0.016443	30.3
30 hrs (108,056 sec)	158.7	159.6	4.697	0.016393	32.8

Figure 2.2-1
CS Pump Allowable Operating Periods @ NPSHa Specified



5.0 Conclusion

The limiting NPSH available for the RHR and CS pumps have been determined for the respective design flow rates during the limiting torus temperature transient. In both cases, the limiting NPSHa falls above the respective allowable NPSH for all times. Therefore, the pumps' operability are not compromised.

Attachment – Letter, Sulzer-Bingham to D. E. Yasi, VYNPC, March 26, 1999

Telefax

SULZER PUMPS
Division of Sulzer Roteq

Sulzer Bingham Pumps Inc.
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Date: 26 March, 1999

To: Mr. Dan Yasi
VERMONT YANKEE DESIGN ENGINEERING OFFICE
Massachusetts

Fax: 8-1-978-568-3732

Pages: 1 (including this one)

Subject: Serial #270839/842 – 280418/419
F-97-10782 30P59
Yankee PO QA42125

With reference to your letter of 12 March 99, based on original NPSHr curve, test data the period of operation will nearly equal the 3000 gpm curve.

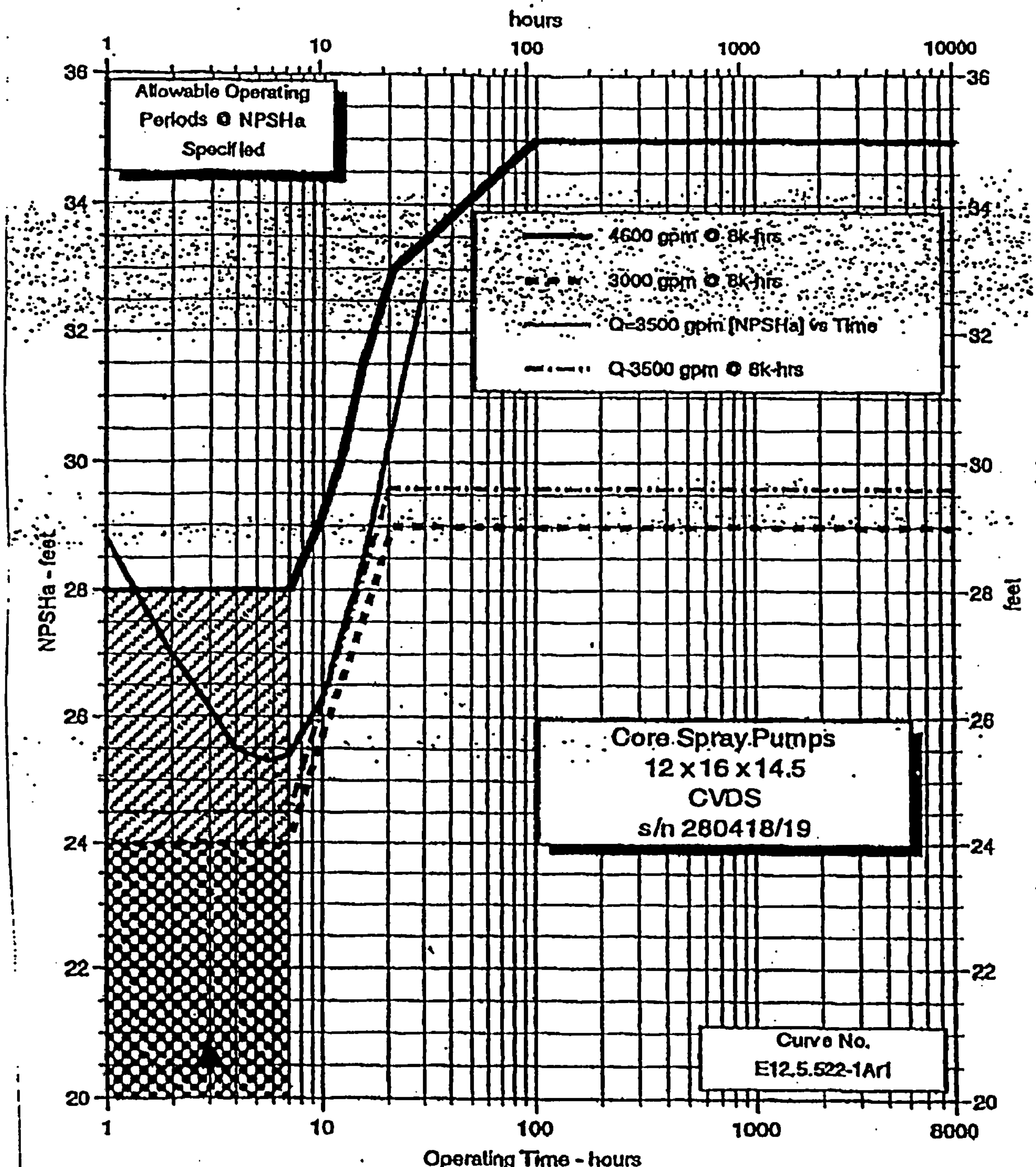
Please see curve (E12.5.522-1Ar1) attached.

Kind Regards,

Kenny Thomson

Kenny Thomson

kta


hours
1
10
100
1000
10000
Allowable Operating Periods @ NPSHa Specified
4600 gpm @ 8k-hrs
3000 gpm @ 8k-hrs
Q=3500 gpm [NPSHa] vs Time
Q-3500 gpm @ 8k-hrs
Core Spray Pumps
12 x 16 x 14.5
CVDS
s/n 280418/19
Curve No.
E12.5.522-1Ar1
NPSHa - feet
feet
36
34
32
30
28
26
24
22
20
1
10
100
1000
8000
Operating Time - hours

Vermont Yankee – Strainer/Torus Design Change Overview
Refuel Outage 20, Spring 1998

Hydraulic Design of the RHR and CS Strainer Assemblies

- Increase in Filtering Area

		Existing	Replacement
RHR	Suction (A or B)	47 ft^2	808 ft^2
CS	Suction (A or B)	24 ft^2	432 ft^2

Note: Areas shown are for each suction line.

- Improved fitting geometry. Replace fabricated Tees with Rams Head, (24" SR Elbows) and Reducing Elbow (24x16 LR Elbow)

- Cylindrical perforated plate strainers replaced with PCI stacked disk strainers with ported core tube for uniform flow distribution to screen area. Strainers have 47" OD disks and 24" OD core tubes.

Predicted Performance Under URG Debris Loads

Conditions Shown are for Strainer Design not Plant Design Basis.		1 RHR-Pump, full flow, LT	2 RHR-Pumps, full flow, ST	2 RHR-Pumps, full flow, LT	1 CS-Pump, full flow, ST	1-CS-Pump, red. flow, LT
Flow Rate per Penetration	gpm	7400	14200	14200	4600	4000
Max Guaranteed Head Loss: Clean Strainer+Fittings	ft H2O	0.35	1.22	1.22	0.55	0.42
Max Guaranteed Head Loss: Strainer+Fittings+Debris	ft H2O	0.56	1.53	1.59	0.69	0.65
Debris Head Loss Alone**	ft H2O	0.21	0.31	0.37	0.14	0.23
** Note: Debris head loss being confirmed by test **Debris quantities based on URG methodology						

SULZER BINGHAM PUMPS INC.

QUALITY LEVEL
☐ Direct
☐ Indirect

SULZER BINGHAM PUMPS INC. DOCUMENT
E12.5.561
NPSH / MINIMUM FLOW STUDY - SUMMARY REPORT
F-97-10782 (30P59)

SALES ORDER 270839/72 and 280418/9

ASME CODE SECTION
CLASS NO.
CODE EDITION (YEAR)
ADDENDA: SEASON
YEAR

CUSTOMER: Yankee Atomic Electric Co. / Vermont Yankee

PROJECT: NPSH Study of RHR CS Pumps

CUSTOMER P.O. NO.: QA42125

CHANGE ORDER NO.

SPECIFICATION NO.: N/A

REVISION NO.

CUSTOMER APPROVAL NUMBER

SPACE FOR CUSTOMER APPROVAL STAMP
(when applicable/available)

CALCULATION VYC- 808-R6
ATTACHMENT 5
PAGE 1 OF 19

CUSTOMER APPROVAL REQUIREMENT

☒ YES ☐ NO ☐ INFORMATION ONLY

CERTIFIED AS A VALID SULZER BINGHAM PUMPS INC. DOCUMENT

☒ For Outside Vendor
☐ For Manufacture at Sulzer Bingham Pumps Inc.
☐ Risk Release Inspection Report # ______
☐ Other (specify) ______

INITIAL APPROVALS (SIGNATURE)		Date
Engrg. Assurance	[signature]	5/26/98
Quality Assurance		

CERTIFICATION (when applicable)

This Document is certified to be in compliance with THE APPLICABLE PURCHASE ORDER, SPECIFICATIONS, PROCEDURES, AND ADDITIONAL REQUIREMENTS LISTED IN THE APPENDICES.

____
Professional Engineer

______ ________
Registration No.

Date ______ (Seal)

Originating Dept.: HQ Engineering

By: Rolf Lueneberg

Title: Hydraulic Design Consultant

Initial Date: 1 May, 1998

APPLICABLE S.O. NUMBERS

270839/842 and 280418/9

Control	Order Number	Rev.
		1

DOCUMENT IDENTIFICATION

Kenny/misc/f97-10782c

CALCULATION VYC- 808 R6
ATTACHMENT 5
PAGE 2 OF 19

SULZER PUMPS
Division of Sulzer Roteq

Status Sheet For: DOCUMENT REVISOIN RECORD

Revision No.	Issue Date	ECN No.	Location	Description	By/Approved
1	05-20-98	---	Page 3 Graphs	Item 3, Added Paragraph Revised graphs to E12.5.522 1B and 2B	KT / DZE 5/20/98

Initial Issue Date ________ Status Sheet Pg. ________ Cont. on Pg. ________ Doc. I.D. ________

SUBJECT: NPSH/Minimum Flow - Study (Summary Report)
Yankee Atomic Electric Company
Vermont Yankee
F97-10782
FS Reference - 30P59
Detailed Report - See E12.5.522

CALCULATION VYC- 808R6
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ORIGINAL PERFORMANCE CURVE NO'S.:

I) 16 x 18 x 26 CVIC: No. 28567 (S.O. No. 270839)
27922 (S.O. No. 270840)
28469 (S.O. No. 270841)
28470 (S.O. No. 270842)

SERVICE: Residual Heat Removal (RHR)

II) 12 x 16 x 14½ CVDS: No. 27691 (S.O. No. 280418)
27692 (S.O. No. 280419)

SERVICE: Core Spray (CS)

I) Introduction

Purpose: To determine the pump operability requirements for (a) minimum pump flow and (b) minimum NPSH for the hypothetical design basis accident mode of operation. The minimum NPSH requirements are based on original pump test data and the application of SBPI knowledge and technology to supplement that data and to extend the range of the original data to higher and/or lower flow rates.

Note: This is a summary report of detailed NPSH report E12.5.522

II) Minimum Flow

A. Expected modes of operation under minimum flow conditions (defined by Vermont Yankee).

RHR - Pumps

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0 to ≤ 4 hours at 350 GPM
≥ 4 hours at 2700 GPM

CS - Pumps

0 to ≤ 4 hours at 300 GPM
≥ 4 hours at 1250 GPM

B. Vibration Data at Minimum Flow (supplied by Vermont Yankee).

Following vibration data were supplied to SBPI:

Data for pump and motor in table - form on 14 January 1998, for:
RHR - Pumps at 425 and 6500 GPM for
CS - Pumps at 300 and 3000 GPM

Additional vibration signatures for pump and motor on 18 March 1998, for CS - pumps at 300 GPM, to complete data from 14 January, 1998.

C. Evaluation of Vibration Data

RHR - Pumps

Data from 14 January 1998 are acceptable for the expected modes of operation under minimum flow conditions, although the overall vibration velocities (in/sec) peak readings were taken at a minimum flow of 425 GPM in lieu of 350 GPM.

CS - Pumps

Data from 14 January 1998 were not complete and indicated signs of unacceptability. Additional vibration signatures from 18 March 1998 are acceptable for the expected modes of operation under minimum flow conditions.

D. Basis for Minimum Flow Requirements

Continuous minimum flow is a function of pump specific speed "N_S", head per stage and suction specific speed N_{SS}-3% (at B.E.P.).

Lower minimum flows than continuous minimum flow are possible and acceptable for shorter durations of operation, depending on acceptable vibration levels (on pump and motor) and NPSH-Margin (NPSHA vs. NPSHR-3%).

For the expected modes of operation under minimum flow conditions:

RHR - Pumps

NPSHA from Vermont Yankee NPSHA-Curve. NPSHR-3% from SBPI Curve No. Id.

At 350 GPM
NPSHA = 35.8 Feet @155 °F, old strainer
NPSHR-3% = 30 Feet

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At 2700 GPM
NPSHA = 34.5 Feet @155 °F, old strai[ner]
NPSHR-3% = 26 Feet

CS - Pumps

NPSHA from Vermont Yankee NPSHA-Curve. NPSHR-3% from SBPI Curve No. IId.

At 300 GPM
NPSHA = 36 Feet
NPSHR-3% = 32.5 Feet

At 1250 GPM
NPSHA = 35.5 Feet
NPSHR-3% = 27 Feet

E. Recommended Minimum Flow Requirements

The recommended minimum flow modes are the same as the expected modes of operations.

RHR - Pumps

0 to ≤ 4 hours at 350 GPM
≥ 4 hours at 2700 GPM

CS - Pumps

0 to ≤ 4 hours at 300 GPM
≥ 4 hours at 1250 GPM

III) NPSH

A. Expected modes of operation under minimum NPSH conditions (defined by Vermont Yankee).

Based on the operating conditions and the NPSHA per May-Witt Decay heat diagrams the expected modes are as follows:

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RHR - Pumps at 7000 GPM

7 hours with NPSHA of 23 to 24 ft.

Plus 5 additional hours with NPSHA of 24 to 26 ft.
Plus 5.5 additional hours with NPSHA of 26 to 28 ft.
Plus 3.5 additional hours with NPSHA of 28 to 29 ft.

CS - Pumps at 3000 GPM

7 hours with NPSHA of 24 to 25 ft.

Plus 2.5 additional hours with NPSHA of 25 to 26 ft.
Plus 2.5 additional hours with NPSHA of 26 to 27 ft.
Plus 3 additional hours with NPSHA of 27 to 28 ft.
Plus 6 additional hours with NPSHA of 28 to 30 ft.

B. Discuss pump performance based on original test data (included original data and curves)

The RHR - and CS - pumps have been NPSH-tested over a limited flow-range. No head-drop was specified on the original curves.

RHR-Pumps: B.E.P. - Flow at 6200 GPM

The most complete NPSH-Test was performed on Pump No. 270840 at maximum impeller diameter of 26.5 in.. NPSH-tests were performed at 6300, 8065 and 9502 GPM (See T-270840-A). 5 to 8 tests points were taken at each of the above capacities to establish the slope and shape of NPSH vs. Head. Purpose of the 'witness' tests were to demonstrate that the pump met the contractual requirements.

Witness - tests for each pump with a trimmed impeller diameter of 25.563 inch, only 2 to 3 NPSH - tests points were taken at capacities of approximately 6300, 7200, 8500 and 8900 GPM.

These tests are not complete enough to determine the exact NPSH-characteristics of the pumps. The duration of the witness - test of each pump, including flow from 0 to runout, pressures, head, RPM, efficiency, power and NPSH took between 1 and 2 hours.

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This implies that the pumps were running only a few minutes with reduced NPSH. This is sufficient time to observe pump behavior at reduced NPSHA. In addition, no vibration readings were taken during these short duration NPSHA tests. A more thorough representation of the complete NPSHR characteristic is test T-270840-A. See SBPI Curve Ic. The difference in NPSHR due to impeller trim does not have a significant influence with these pumps.

CS-Pumps: B.E.P. - Flow at 3750 GPM

The most representative NPSH-Test was performed during Test No. 176101 at 13.81 inches and 13.00 inches impeller diameters. It was a pump for a different order, but an identical pump. NPSH - tests were performed at approximately 1780 RPM.

Converted to 3582 RPM, by using the affinity laws, the flow rates were 3005, 4037, 5038, 5120, 6000, 6020 and 6524 GPM (see T-176101-D/G). 4 to 10 tests points were taken at each of the above capabilities to establish the slope and shape of NPSH vs. Head. These tests are sufficient to develop NPSHR characteristics for the pump and are representative of the units delivered on the above serial numbers. Trim diameters have been factored in the developed NPSHR curves.

The most complete and representative test T-176101-D/G. (see also SBPI Curve No. IIc).

1. **Relationship to "Knee" of Pump Curve**

   When plotting the results of an NPSH-test (NPSH vs. Head), starting with ample NPSH, the head will either stay constant, vary or drop slightly with reducing NPSH. At some reduced NPSH value, the head will fall off more quickly before falling off totally. This defines the "Knee" of an NPSH-test.

   The knee may be very sharp, that means 1%, 3%, 6%, etc. head-drop will occur at about the same NPSHA value. Operation near or close to this type of knee is not recommended. The knee may also be well-rounded, that means 1%, 3%, 6% etc. head-drop will happen at different NPSHA values. To develop the shape of the NPSHR knee several test points are required.

2. **Similarity to Other Pumps Used in Nuclear Application**

   Pump designs provided for the above services are found in other nuclear installations in the same or similar applications. They are basically of similar style and design, but may differ in nozzle and maximum impeller sizes. There are pumps of same specific speed, suction specific speed and impeller inlet design features (NPSH).

3. **Relevant Operating Experience of Similar Pumps at Minimum NPSH**

   Operating conditions at various nuclear stations vary, however similar units to those furnished have been supplied to other installations with similar reduced NPSH levels during a nuclear incident. Similar reviews have been conducted for them.

Specified "normal" operating conditions (NPSHA) are not that close to NPSHR-3% or NPSHR-6%.

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Operating for short durations at NPSHR-3% to NPSHR-6% should not be detrimental to the pump life in this service.

4. **Cavitation-Tests performed on same or similar pumps and conclusion from those tests.**

Cavitation (NPSH) - Tests have been performed on same pumps or similar pumps that have been used on the NPSH-study for Vermont Yankee.

NPSH-Test on same pumps is T-270840-A (RHR-Pump) and T-176101-D/G (CS-Pump) for discussion and conclusion see IIIB.

NPSH - Tests on similar pumps are used to establish tendencies and extrapolation of NPSH-Curves. (See SBPI Curve No. Id & IId). Similar pumps are of same suction specific speed, number of vanes and suction vane inlet angles.

5. **Acceptability of the units in their specified services.**

RHR - Pumps:

When operating for seven (7) hours at 7000 GPM with NPSHA of 23 to 24 feet, the pumps will be in the cavitation mode. The head-drop will be above 6% but the NPSHA is still greater than the original minimum operational NPSH (See SBPI Curve No. Ic and Id). The pumps, if operated with the minimum NPSH, are within acceptable limits of the NPSH "knee".

The pumps will remain acceptable following the "Postulated Accident Scenario" and operation under reduce NPSH conditions, providing NPSHA > NPSHR-3%.

CS - Pumps:

When operating for seven (7) hours at 3000 GPM with NPSHA of 24 to 25 feet, the pumps will be in the cavitation mode. The head-drop however will be less than 3% (see SBPI Curve No. IIc and IId). The pumps, if operated with the minimum NPSH limits, have adequate margin prior to the NPSH "knee".

The pumps will remain acceptable following the "Postulated Accident Scenario" and operation under minimum NPSH conditions, providing NPSHA > NPSHR-3%.

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C. Extrapolation to higher/lower flows using test date from other sources.

1. Technical Basis for Extrapolation

When pumps have been NPSH-tested for a small flow-range only (Vermont Yankee) and NPSH-data are required outside this flow-range, the NPSH-curves have to be extrapolated. Only NPSH-tests of pumps of similar style, design, specific speed, suction specific speed, impeller number of vanes and suction vane angles can be used for this purpose.

2. Pump data selected and similarity to Vermont Yankee pumps

RHR - Pumps:

Following pump sizes have been used to extend NPSHR to lower flows (see SBPI curve No. Ia):

18 x 24 x 28 CVIC
8 x 10 x 21 CVIC

CS - Pumps:

Following pump sizes have been used to extend NPSHR to lower flows (see SBPI curve No. IIa):

12 x 14 x 14$^1/_2$ CVDS
14 x 16 x 23 CVDS

3. Predicted NPSH at lower/higher flow rates as extrapolations of original test data

In this case, the minimum flow rates are extremely low:

RHR-Pumps: 350 GPM $\hat{=} \dfrac{350}{6200} x 100 = 5.6\%$ of B.E.P. Flow

CS-Pumps: 300 GPM $\hat{=} \dfrac{300}{3750} x 100 = 8.0\%$ of B.E.P. Flow

No NPSH-tests of same or similar pumps are available at these low percentages of B.E.P. flow.

Extrapolation to these low flow rates based only on estimation and experience of NPSH-tests on other style of pumps. Experience comes from NPSH-tests which have been performed in recent years, when more detailed NPSH-tests were required.

Extrapolation for higher flow rate NPSHR is not necessary since sufficient test data exists for these flow rates. If extrapolation for higher flow rates is necessary a similar method will be used.

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D. **Basis for Minimum NPSH Recommendations**

1. **No Permanent Pump Damage Due to Cavitation**

Depending on water temperature and water chemistry there can be some 'frosting' (e.g. light pitting) on the impeller suction vanes, but there will be no detrimental pump damage due to cavitation when operating at minimum NPSH for the specified hours of operation.

This applies mainly to the RHR-pumps when operating for seven (7) hours at 7000 GPM with NPSHA of 23 to 24 feet. It will apply to a lesser degree to the CS-pumps when operating for seven (7) hours at 3000 GPM with NPSHA of 24 to 25 feet.

2. **Operation above the "Knee" of the Pump Curve**

Maintaining the minimum NPSH values is a "must" when operating near or at the NPSHR knee. For continuous operation this is essential, since small variances in product temperature can suddenly reduce the NPSHA. Provided the NPSH values are supplied for the RHR and CS services and durations, at these values limited, operation at the NPSHR "knees" are acceptable.

Short-Term Operation at the "Knee" is acceptable providing temperature is controlled.

3. **Conformance to Original Pump Requirements and Extrapolated Requirements, as defined herein**

These pumps meet the original NPSHR requirements as specified. The original pump NPSH requirements were not well defined. The result was only two (2) NPSH-Test points for each capacity were measured. From two (2) NPSH-test points it is not possible to establish the "knee". At each NPSH-test point (during witness tests) the pumps were operating only a few minutes and the capacity-range was limited. This was not considered critical since similar pumps of the same hydraulics had been comprehensively tested.

The extrapolated NPSH requirements apply mainly to a flow regime of 350 and 2700 GPM for the RHR-pumps and 300 and 1250 GPM for the CS-pumps as described under III.c.3. Due to unknown suction vane profile and clean-up on NPSH-margin as described under II.d is required.

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E. Recommended Minimum NPSH Requirements

1. Acceptable durations of operation:

RHR-Pumps:

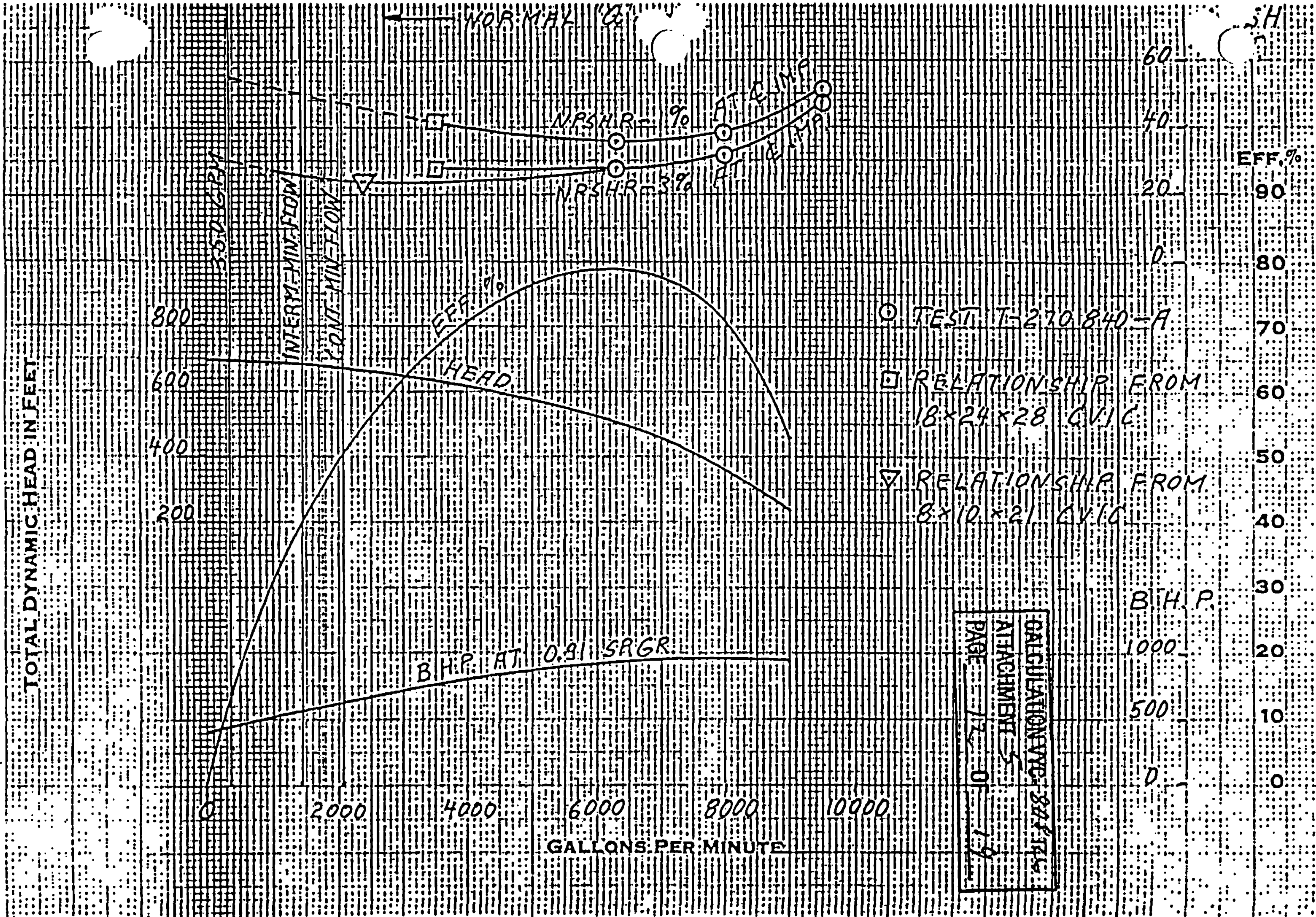
At minimum flow of 350 and 2700 GPM: as described under II.d and II.e.
At 7000 GPM: as shown on SBPI Curve No. E12.5.522-2B

CS-Pumps:

At minimum flow of 300 and 1250 GPM: as described under II.d and II.e.
At 3000 and 4600GPM: as shown on SBPI Curve No. E12.5.522-1B

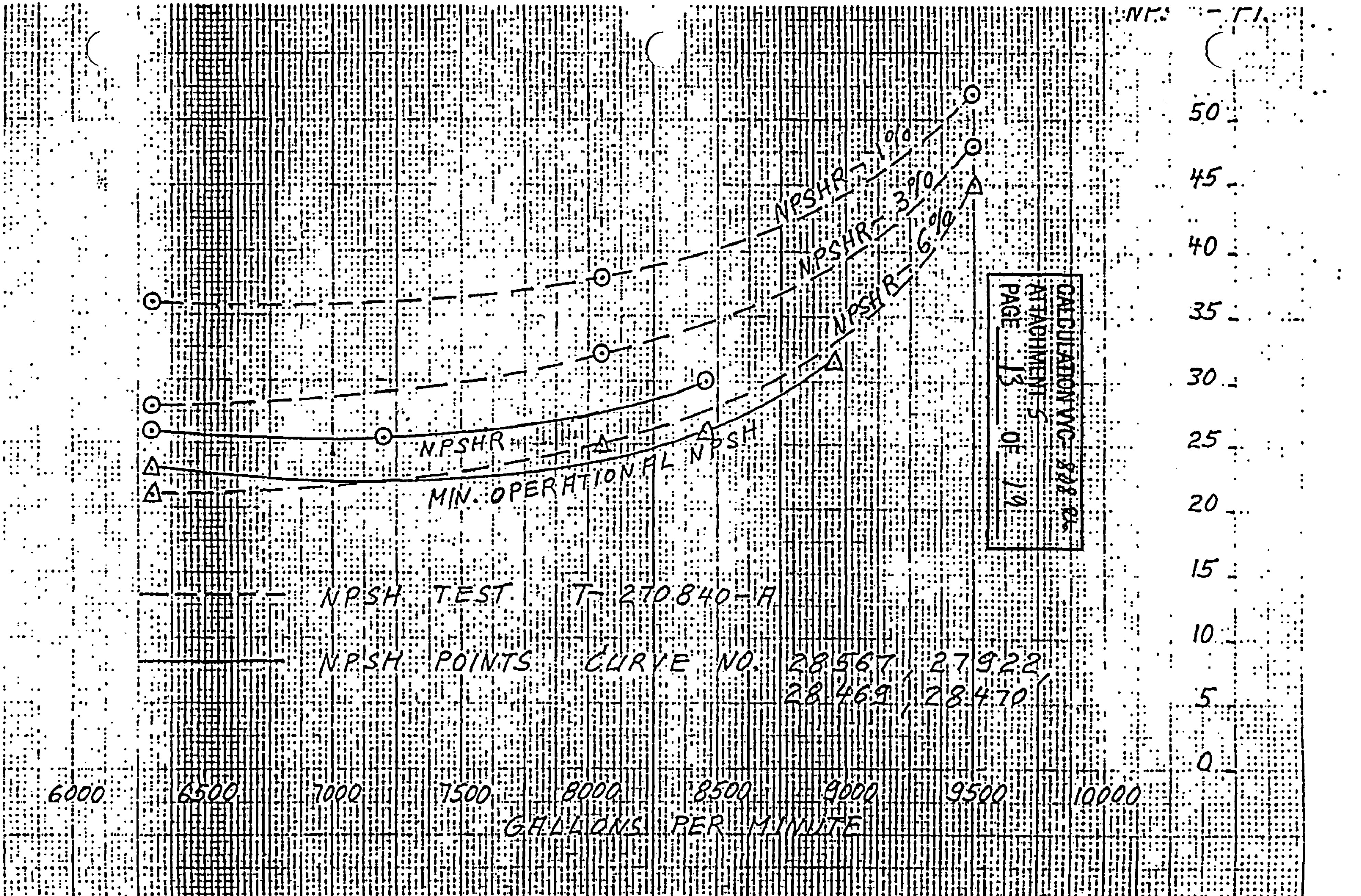
2. Purpose of this report:

This report and review was conducted to clarify test results taken approximately thirty (30) years earlier. It is also intended to provide additional understanding regarding the limits of these machines both hydraulically and mechanically. These machines are suitable for the services they were originally supplied to, however they must operated within the agreed limits.



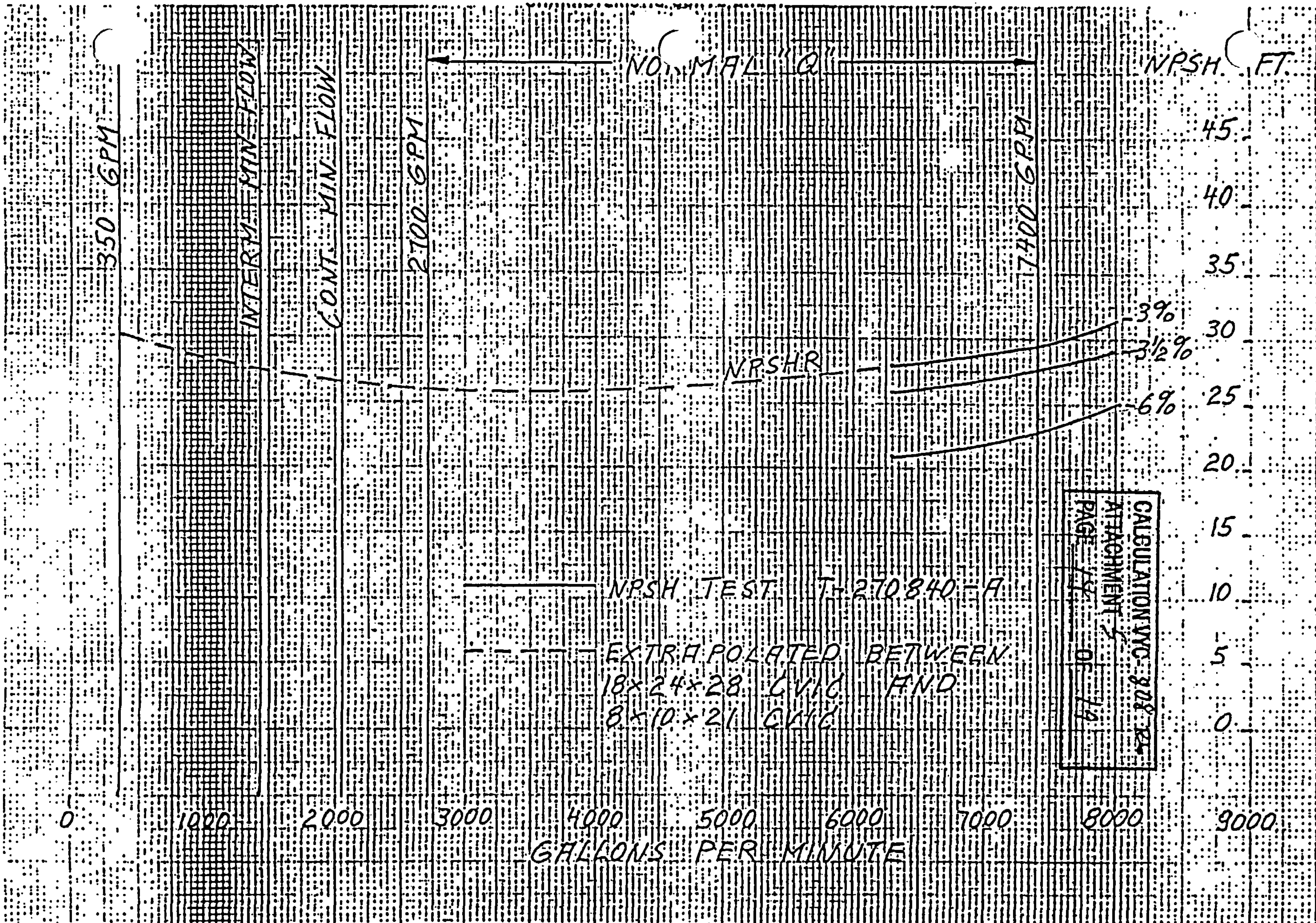
YANKEE ATOMIC ELECTRIC VERMONT YANKEE RHR PUMPS S.O. NO. 270839/842	PUMP ENGINEERING DEPT. **SULZER BINGHAM PUMPS INC.** 2. Dec. 97; R.L.	IMPELLER MAX. DIA. 26.500"	16 × 18 × 26 CVIC PUMP		
		MIN. DIA.	DIA. IMPELLER 25.563"	IMPELLER PATT. 1613 CVIC-1	1785 R.P.M.
		EYE AREA 121.3 SQ. IN.	N.P.S.H. REQUIRED	REFERENCE	CURVE NO. Ia

Form No. 930-1



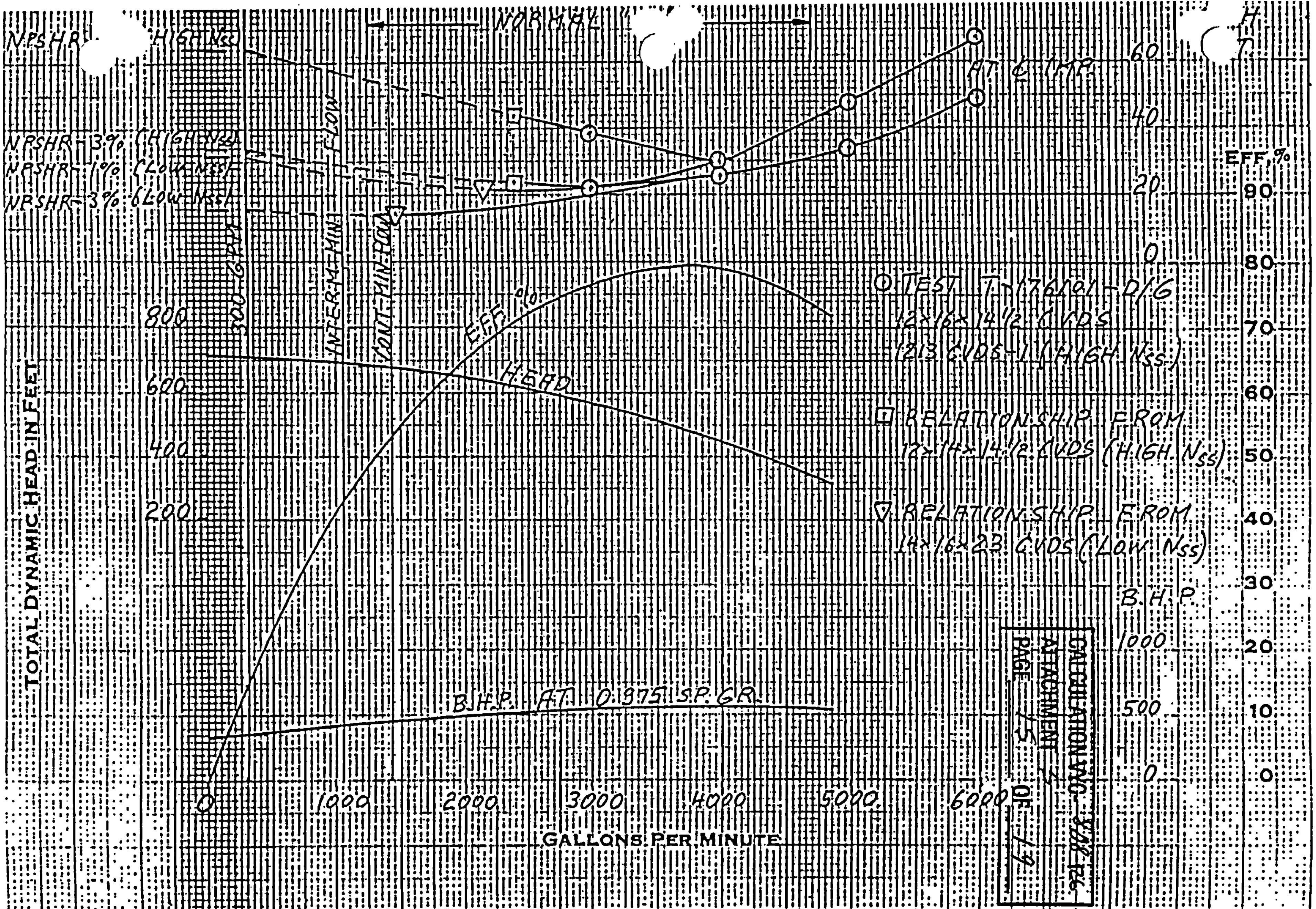
YANKEE ATOMIC ELECTRIC VERMONT YANKEE RHR PUMPS S.O. NO. 270 839/842	PUMP ENGINEERING DEPT. **SULZER BINGHAM PUMPS INC.** 18 MAR 98; R.L.	IMPELLER MAX. DIA. 26.500"	16×18×26 CVIC PUMP		
		MIN. DIA.	DIA. IMPELLER 25.563"	IMPELLER PATT. 1613 CVIC-1	1785 R.P.M.
		EYE AREA 121.3 SQ. IN.	N.P.S.H. REQUIRED	REFERENCE	CURVE NO. I C

Form No. 308-2



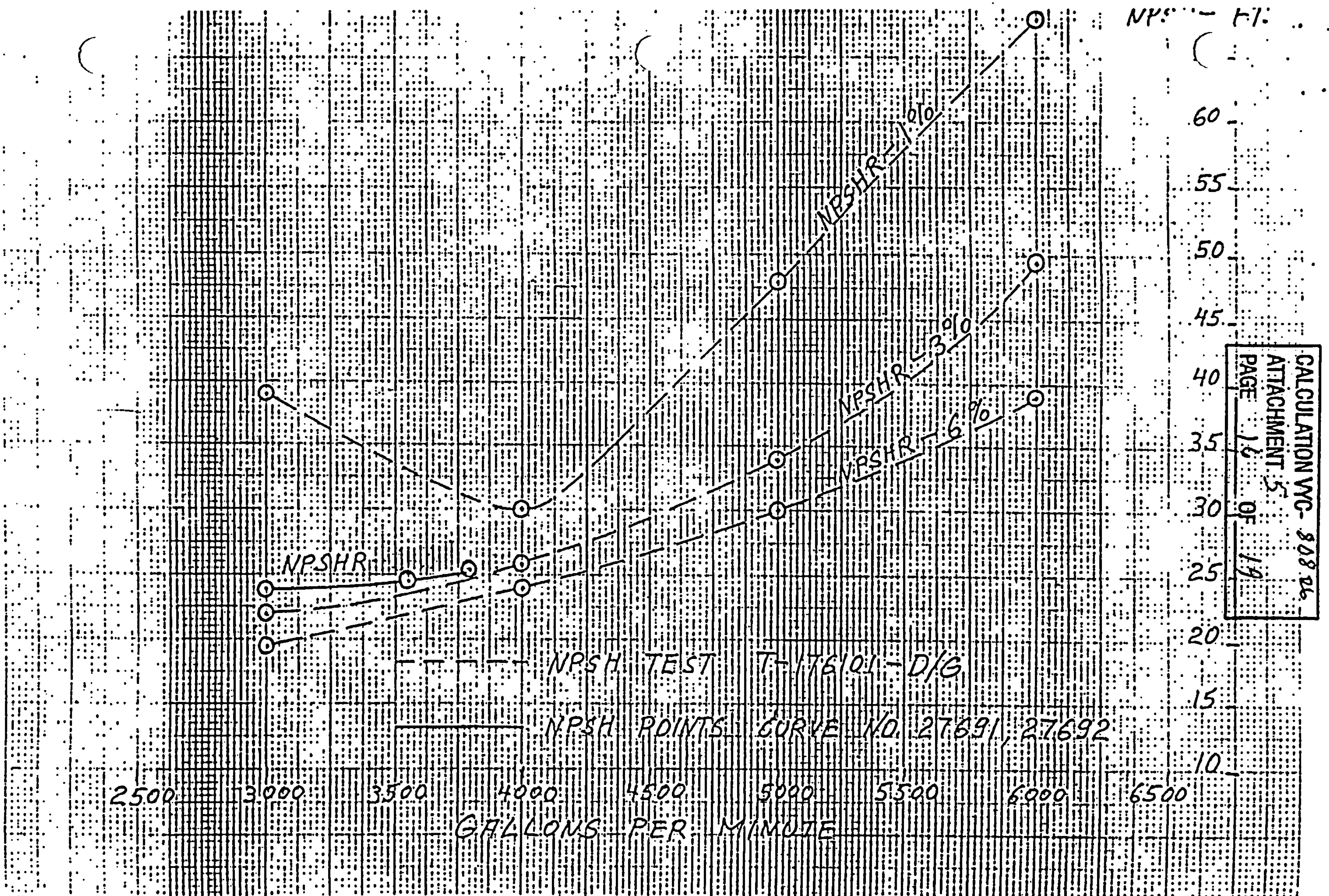
YANKEE ATOMIC ELECTRIC VERMONT YANKEE R.H.R. PUMPS S.O. NO. 270 839/842	PUMP ENGINEERING DEPT. **SULZER BINGHAM PUMPS INC.** 16. JAN. 98, R.L.	IMPELLER MAX. DIA. 26.500"	16×18×26 CVIC PUMP		
		MIN. DIA.	DIA. IMPELLER 25.563"	IMPELLER PATT. 1613 CVIC-1	1785 R.P.M.
		EYE AREA 121.3 SQ. IN.	N.P.S.H. REQUIRED	REFERENCE Ia, b, c	CURVE No. Id

Form No. 308-2



YANKEE ATOMIC ELECTRIC VERMONT YANKEE CS PUMPS S.O. NO. 280418/419	PUMP ENGINEERING DEPT. **SULZER BINGHAM PUMPS INC.** 4.DEC.97 ; R.L.	IMPELLER MAX. DIA. 14.500"	12×16×14½ CVDS PUMP	
		MIN. DIA.	DIA. IMPELLER 12.500"	IMPELLER PATT. 1213 CVDS-1	3582 R.P.M.
		EYE AREA 72.2 SQ. IN.	N.P.S.H. REQUIRED	REFERENCE	CURVE NO. IIa

Form No. 930-1



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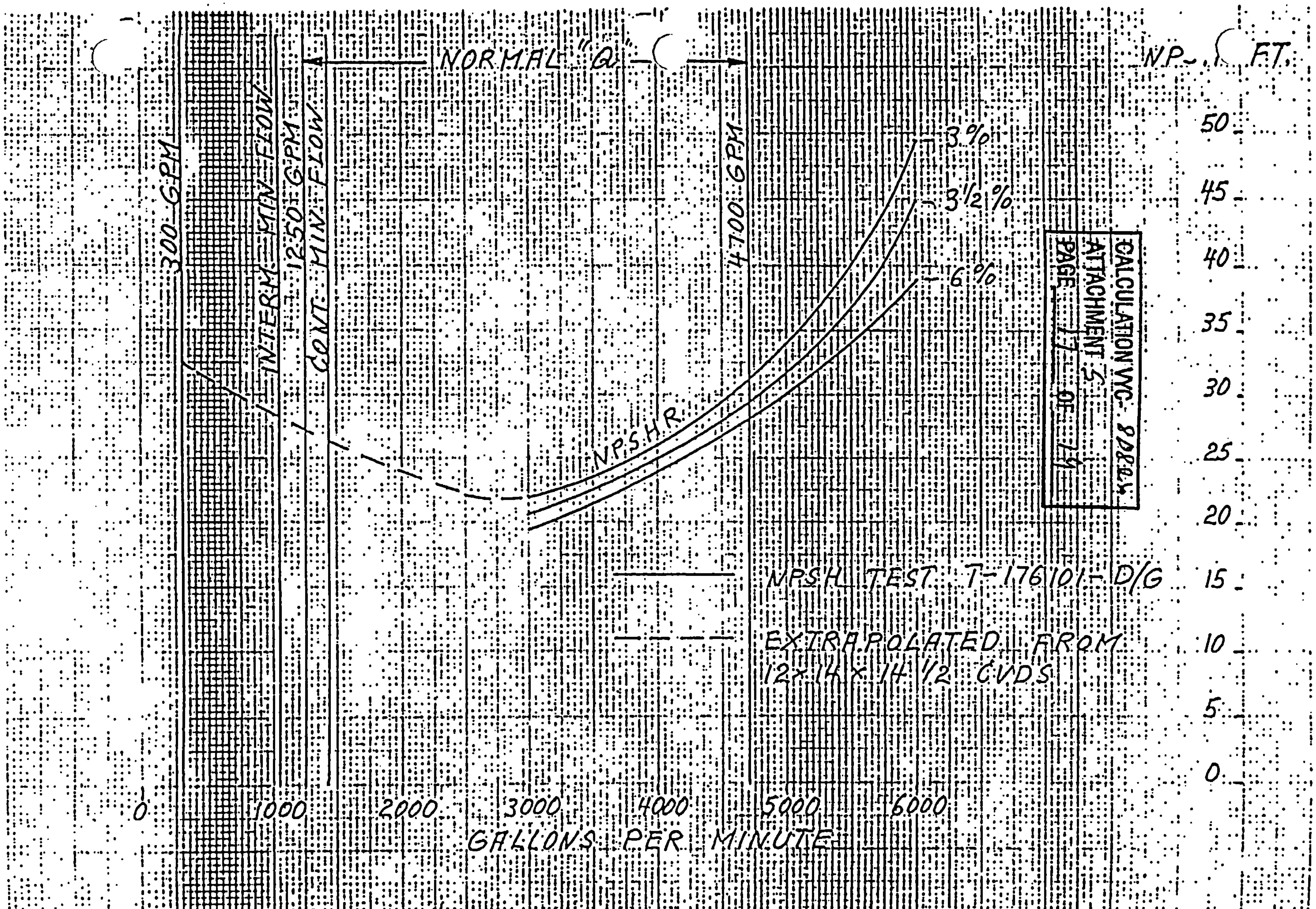
YANKEE ATOMIC ELECTRIC VERMONT YANKEE CS PUMPS S.O. NO. 280-418/419	PUMP ENGINEERING DEPT. **SULZER BINGHAM PUMPS INC.** 18 MAR 98; R.L.	IMPELLER MAX. DIA. 14.500" MIN. DIA. EYE AREA 72.2 SQ. IN.	12 × 16 × 14 1/2 CVDS PUMP		
			DIA. IMPELLER 12.500"	IMPELLER PATT. 1213 CVDS-1	3582 R.P.M.
			N.P.S.H. REQUIRED	REFERENCE	CURVE NO. II C

Form No. 208-2



YANKEE ATOMIC ELECTRIC VERMONT YANKEE C.S. PUMPS S.O. NO. 280 418/419	PUMP ENGINEERING DEPT. **SULZER BINGHAM PUMPS INC.** 16. JAN. 98; R.L.	IMPELLER MAX. DIA. 14.500"	12×16×14 1/2 CVDS PUMP	
		MIN. DIA.	DIA. IMPELLER 12.500"	IMPELLER PATT. 1213 CVDS-1 / 3582 R.P.M.
		EYE AREA 72.2 SQ. IN.	N.P.S.H. REQUIRED	REFERENCE II a,b,c / CURVE No. II d

Form No. 308-2

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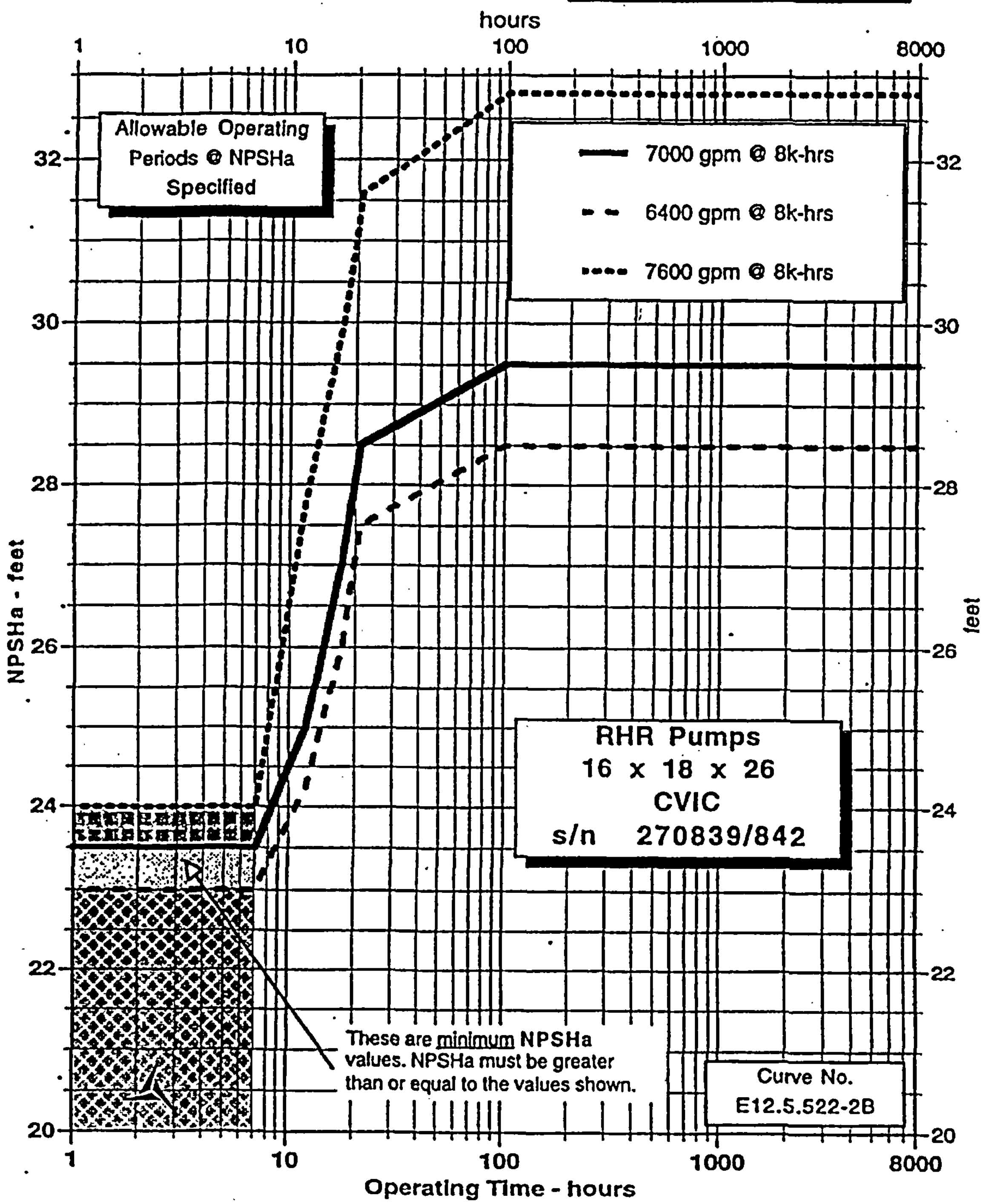

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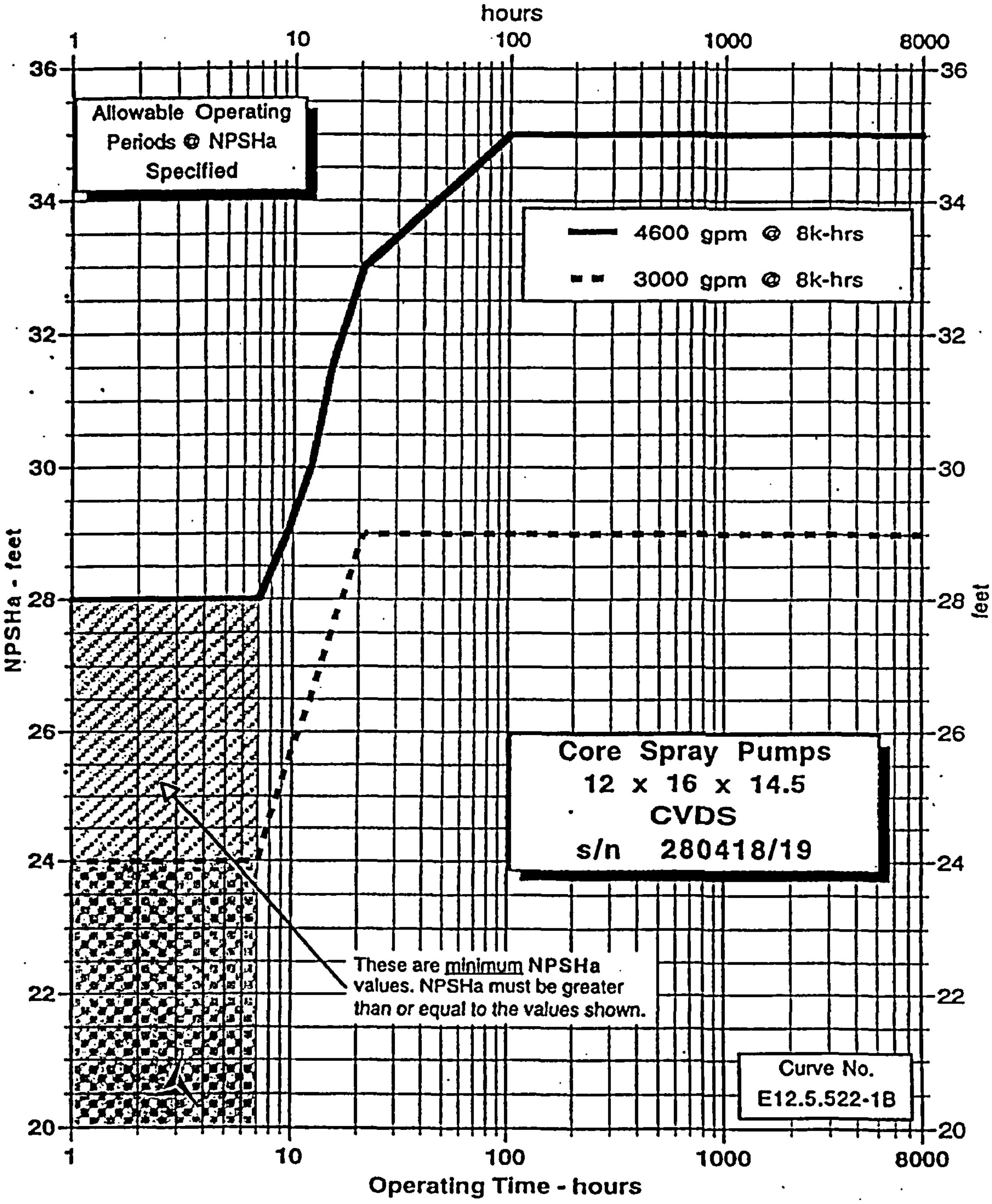

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SUBJECT Core Spray NPSH Evaluation

PREPARED BY ____ DATE ____ REVIEWED BY ____ DATE ____ WORK ORDER NO. 4922

5.0 References:

~~(a) Drawing, G-191168, Flow Diagram - Core Spray System~~ StH · BLS 4/25/00

(b) Drawing, 5920-9209, Core Spray (CS) Part 3

(c) Technical Paper No. 410, Flow of Fluids Through Valves Fittings and Pipe, Crane Co., 24th Printing, 1988

(d) Goulds Pump Manual, Gould Pumps Inc., Seneca Falls N.Y., 1973

(e) Pump Curve for Core Spray Pumps #280418,#280419, Bingham Pump Co., Curve Nos. 27691, 27692

(f) Drawing (CB & I), 6202-233, Torus Penetrations

(g) VYNPS FSAR, Fig. 5.2-1

(h) Drawing, G-191206, Core Spray System Piping Plan

(i) Drawing, G-191207, Core Spray System Piping Sections

(j) "Thermodynamic Properties of Steam", Keenan and Keyes, John Wiley and Sons, New York, 1959

(k) Drawing, 5920-6683, C.S. Suction Strainer for Torus Penetrations X-226A and X-226B

~~(l) Handbook of Hydraulic Resistance, I.E.Idelchik, 2nd Ed., Hemisphere Publishing Co., New York, 1986~~

(m) Hydraulic Institute Engineering Data Book, 1st Ed.

6.0 Calculation:

6.1 Suction Piping Lengths

Each Core Spray pump takes suction from its own Torus penetration. The two suction piping paths are essentially "mirror images".

6.1.1 Torus to CS Pump piping:

Pipe size = 12" STD [Ref.(a)]

Piping Lengths [Ref.(b)]

	Feet	Inches	
	3	0.0	
	3	2.5	
	3	1.5	
	1	9.0	
	7	6.0	
	4	3.0	
	0	3.5	
	1	0.5	
	1	6.0	
	2	4.0	
	3	6.0	
	2	5.0	
	0	11.5	
Total	30	58.5	or 34.875', say 35'

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SUBJECT Core Spray NPSH Evaluation

PREPARED BY ______ DATE ______ REVIEWED BY ______ DATE ______ WORK ORDER NO. 4922

6.1.2 Fittings [Ref. (b)]

	L/D		[Ref. (c)]
90° LR 16"x12" Red. Elbow	14	(Note 1)	
45° LR Elbow	11		[Ref. (m)]
67 1/2° LR Elbow	14	(Note 1)	
90° LR Elbow	14		
90° LR Elbow	14		
TEE (Str. Run)	20		
16"x12" Reducer (Exp.)	7		
Valve, Gate (CS-7A)	8		
~~Strainer Entrance Tee (16")~~	~~30~~		~~[Ref. (l)]~~
Total	~~132~~ 102		

Total equivalent length (12") pipe = 35 + ~~132~~ 102 = ~~167"~~ ~~135~~ 137 BCB 4/25/00

Note 1: Conservatively assumed same as 90° LR Elbow.

6.1.3 Correction to Sched. 40 [Ref.(c)]

12" STD : I.D. = 12.00"

12" SCH 40 : I.D. = 11.938"

$dP_a = dP_{40} (D_{40}^5 / D_a^5)$ [Ref.(c), @ B-15]

Therefore,

$dP_{STD} = dP_{40} (11.938^5 / 12.00^5) = dP_{40} (0.974)$

Equivalent length Sched. 40:

$L_{40} = L_{STD} (0.974)$

= (~~167~~ 137)(0.974) = ~~162.63, say 163'~~ 133

6.2 Friction loss:

$H_f = (L) (h_f)$ [Ref.(c)]

@ 3000 gpm, h_f = 0.731 psi/100', (60 °F water)

$H_{f@3000}$ = (~~163~~ 133 ft.)(0.731 psi/100 ft.)(2.31 ft./psi) = ~~2.752~~ 2.25 ft.

For other flow rates:

$H_{f1} = H_{f0} (Q_1^2 / Q_0^2)$

Note: This is slightly conservative for $Q_1 > Q_0$, but difference is insignificant in the range of interest.

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SUBJECT Residual Heat Removal (LPCI) NPSH Evaluation

PREPARED BY_____ DATE_____ REVIEWED BY_____ DATE_____ WORK ORDER NO. 4922

5.0 References:

✓ (a) Drawing, G-191172, Flow Diagram - Residual Heat Removal System

✓ (b) Drawing, 5920-9284, Residual Heat Removal (RHR) Part 5.

✓ (c) Technical Paper No. 410, Flow of Fluids Through Valves Fittings and Pipe, Crane Co., 24th Printing, 1988

~~(d) Goulds Pump Manual, Gould Pumps Inc., Seneca Falls N.Y., 1973~~

~~(e) Pump Curve for RHR Pump #270841, Bingham Pump Co., Curve No. 28469~~

~~(f) Drawing (CB & IO, 6202-233, Torus Penetrations~~

~~(g) VYNPS FSAR Fig. 5.2-1~~

~~(h) Drawing, G-191211, RHR System Piping Sections~~

~~(i) Drawing G-191210, RHR System Piping Plan~~

~~(j) "Thermodynamic Properties of Steam", Keenan and Keyes, John Wiley and Sons, New York, 1959~~

✓ (k) Handbook of Hydraulic Resistance, I.E.Idelchik, 2nd Ed., Hemisphere Publishing Co., New York, 1986

✓ (l) Hydraulic Institute Engineering Data Book, 1st Ed.

~~(m) Drawing 5920-6764, RHR Suction Strainer for Torus Penetrations X-224A & X-224B~~

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SUBJECT Residual Heat Removal (LPCI) NPSH Evaluation

PREPARED BY______ DATE______ REVIEWED BY______ DATE______ WORK ORDER NO. 4922

6.0 Calculation:

6.1 Suction Piping Lengths

Two RHR pumps take suction from one Torus penetration therefore a portion of suction piping is common to both pumps from the Torus to a Tee at which point there is a single branch to each pump. Both the A and B RHR loops are mirror image so the calculation need be done for only one loop. (A and C pump loop.) By inspection of the piping isometric drawings it is noted that there are some small differences in the single suction lines the most significant of which is that one has a straight run at the Tee while the other is a branch run. The run with the greater loss will be used for conservative results.

6.1.1 Common piping Torus to Tee 24" pipe.

Pipe size = 24", 0.375 wall. I.D. = 23.25" (STD.) [Ref.(a)]

Piping Lengths [Ref.(b)]

	Feet	Inches		
	1	2.625		
	3	6.125		
	4	0.0		
Total	8	8.75	or	8.73 ft.

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SUBJECT Residual Heat Removal (LPCI) NPSH Evaluation

PREPARED BY________ DATE________ REVIEWED BY________ DATE________ WORK ORDER NO. 4922

6.1.2 Fittings

	L/D		
	L/D		[Ref.(c)]
33° LR Elbow	7	(estimated)	
90° SR Elbow	20		
26"x24" Reducer (Exp.)	1		
~~Strainer Entrance Tee~~	~~119~~		~~[Ref.(h)]~~
~~24"-24° Mitre Bend~~	~~6~~		

Total ~~153~~ 28 or ~~296.44~~ 54.25 ft. equivalent

6.1.3 Correction to Sched. 40 (20")

24" STD : I.D. = 23.25"

20" SCH 40 : I.D. = 18.814"

$dP_a = dP_{40} (D_{40}{}^5 / D_a{}^5)$ [Ref.(c) @ B-15]

Therefore,

$dP_{STD} = dP_{40} (18.814^5 / 23.25^5) = dP_{40} (0.347)$

Equivalent length Sched. 40:

$L_{40} = L_{STD} (0.347)$

$= (8.73 + \cancel{296.44}\ 54.25) (0.347) = \cancel{105.89}\ 21.85$

6.1.4 Common piping Torus to Tee 26" pipe.

Pipe size = 26", 0.375 wall. I.D. = 25.25" [Ref.(a)]

Piping Lengths [Ref.(b)]

	Feet	Inches		
	1	4.125		
	2	8.25		
	17	10.125		
	5	5.5		
	2	2.0		
	6	0.0		
	4	3.0		
	2	0.0		
	1	3.0		
Total	40	36	or	43.00 ft.

6.1.5 Fittings

	L/D		
			[Ref.(c)]
45° LR Elbow	11		[Ref.(1)]
45° LR Elbow	11		[Ref.(1)]
90° SR Elbow	20		
90° SR Elbow	20		
Total	62	or 130.46 ft. equivalent	

6.1.6 Correction to Sched. 40 (20")

26" STD : I.D. = 25.25"

20" SCH 40 : I.D. = 18.814"

$dP_a = dP_{40} \ (D_{40}{}^5 / D_a{}^5)$ [Ref.(c) @ B-15]

Therefore,

$dP_{STD} = dP_{40} \ (18.814^5 / 25.25^5) = dP_{40} \ (0.230)$

Equivalent length Sched. 40:

$L_{40} = L_{STD} \ (0.230)$

$= (43.00 + 130.46) \ (0.230) = 39.90'$

6.1.7 Total common piping 20" Sched. 40 equivalent

$L_{COM} =$ ~~105.89~~ 21.85 + 39.90 = ~~145.79'~~ 61.75 (20", Sch.40)

6.1.8 Single piping Tee to Pump A 26"

Pipe size = 26", 0.375 wall. I.D. = 25.25" . [Ref.(a)]

Piping Lengths [Ref.(b)]

	Feet	Inches		
	3	10.0		
	2	5.5		
	2	3.5		
Total	7	19	or	8.58 ft.

6.1.9 Fittings

	L/D	
Tee (Str. Run)	20	
90^o SR Elbow	20	
10^o LR Elbow	1	(estimated)
26"x20" Reducer	17	
Total	58	or 122.04 ft. equivalent

6.1.10 Correction to Sched. 40 (20")

26" STD : I.D. = 25.25"

20" SCH 40 :I.D. = 18.814"

$dP_a = dP_{40} (D_{40}{}^5 / D_a{}^5)$ [Ref.(c) @ B-15]

Therefore,

$dP_{STD} = dP_{40} (18.814^5 / 25.25^5) = dP_{40} (0.230)$

Equivalent length Sched. 40:

$L_{40} = L_{STD} (0.230)$

$= (8.58 + 122.04) (0.230) = 30.04'$

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6.1.11 Single piping Tee to Pump A 20"

Pipe size = 20", 0.375 wall. I.D. = 19.25" [Ref.(a)]

Piping Lengths [Ref.(b)]

	Feet	Inches		
	1	8.0		
	3	7.0		
	3	5.0		
	2	11.0		
	2	7.0		
	3	5.0		
Total	14	43	or	17.58 ft.

6.1.12 Fittings

	L/D		[Ref.(c)]
90° SR Elbow	20		
Valve (10-13A),Gate	8		
90° SR Elbow	20		
Tee (20x20x20,Str. Run)	20		
Tee (20x4x20,Str. Run)	1		[Ref.(k)]
20"x18" Reducer	2	(estimated)	
Total	71	or 113.90 ft. equivalent	

6.1.13 Correction to Sched. 40 (20")

20" STD : I.D. = 19.25"

20" SCH 40 : I.D. = 18.814"

$dP_a = dP_{40} (D_{40}^{5} / D_a^{5})$ [Ref. (c) @ B-15].

Therefore,

$dP_{STD} = dP_{40} (18.814^{5} / 19.25^{5}) = dP_{40} (0.892)$

Equivalent length Sched. 40:

$L_{40} = L_{STD} (0.892)$

$= (17.58 + 113.90) (0.892) = 117.28'$

6.1.14 Total single piping Tee to Pump A 20" Sched. 40 equivalent

$L_{T-A} = 30.04 + 117.28 = 147.32'$

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6.1.15 Single piping Tee to Pump C 20"

Pipe size = 20", 0.375 wall. I.D. = 19.25" [Ref.(a)]

Piping Lengths [Ref.(b)]

	Feet	Inches	
	2	3.5	
	0	9.5	
	2	10.0	
	3	4.0	
	3	4.0	
	3	0.0	
	1	10.5	
	2	9.0	
Total	16	50.5	or 20.21 ft.

6.1.16 Fittings

	L/D		
Tee (Br. Run)	60		[Ref.(c)]
90° SR Elbow	20		
Valve (10-13C), Gate	8		
90° SR Elbow	20		
Tee (20x20x20, Str. Run)	20		
Tee (20x4x20, Str. Run)	1		[Ref.(k)]
20"x18" Reducer	2	(estimated)	
10° LR Elbow	1		
Total	132	or 211.75 ft. equivalent	

6.1.17 Correction to Sched. 40 (20")

20" STD : I.D. = 19.25"

20" SCH 40 :I.D. = 18.814"

$dP_a = dP_{40} \left(D_{40}^{\ 5} / D_a^{\ 5} \right)$ [Ref.(c) @ B-15]

Therefore,

$dP_{STD} = dP_{40} \left(18.814^{\ 5} / 19.25^{\ 5} \right) = dP_{40} \ (0.892)$

Equivalent length Sched. 40:

$L_{40} = L_{STD} \ (0.892)$

$= (20.21 + 211.75) \ (0.892) = 206.91'$

6.1.18 Total single piping Tee to Pump C 20" Sched. 40 equivalent

$L_{T-C} = 206.91'$

6.1.19 Piping Summary - Equivalent Feet 20" Sched. 40

Torus to Tee (Common), (L_{COM}) = ~~145.79'~~ 61.75

Tee to Pump A, (L_{T-A}) = 147.32'

Tee to Pump C, (L_{T-C}) = 206.91'

Pump C has the longer run ; use for conservative result.

6.2 Friction Loss

$H_f = H_{f common} + H_{f\ single}$

Head loss in common line will depend on number of pumps operating. Assume that if both pumps are operating they are operating at the same flow rate.

Consider two configurations : One pump operation (Case I), and two pump operation (Case II).

VERMONT YANKEE DESIGN ENGINEERING
CALCULATION NO. VYC- 808
REVISION NO. 5
ATTACHMENT 7 PAGE 11 OF 12

YANKEE ATOMIC ELECTRIC COMPANY

CALCULATION NO. ~~VYC-1309~~ PAGE ~~11~~ OF

SUBJECT Residual Heat Removal (LPCI) NPSH Evaluation

PREPARED BY ______ DATE ______ REVIEWED BY ______ DATE ______ WORK ORDER NO. 4922

Case I Friction Loss :

$H_f = H_{f\ common} + H_{f\ single}$

and $H_{fa} = H_{fb} (Q_a^2 / Q_b^2)$ [Ref.(c)]

as a function of flow where $Q_{common} = Q_{single}$ for Case I.

(Note: This is slightly conservative for $Q_a < Q_b$, but difference is not significant in range of interest.)

$H_{fI} = H_{fc0} (Q^2 / Q_0^2) + H_{fs0} (Q^2 / Q_0^2)$

where H_{fc0} = friction loss at Q_0 for the common piping

and H_{fs0} = friction loss at Q_0 for the single piping

For Q_0 = 7000 gpm, friction loss for 20" Sched. 40 pipe is 0.376 psi per 100',

[Ref.(c)]

$H_{fc0} = L_{COM} (H_{f0})$

$= (\cancel{445.39}\ 61.75) (0.376 / 100) (2.31\ ft/psi) = \cancel{1.266'}\ 0.536$

$H_{fs0} = L_{T-C} (H_{f0})$

$= (206.91) (0.376 / 100) (2.31\ ft/psi) = 1.80'$

$H_{f0I} = H_{fc0} + H_{fs0} = \cancel{1.266}\ 0.536 + 1.80 = \cancel{3.066'}$ @ 7000gpm 2.336

therefore,

$H_{fI} = \cancel{3.066}\ 2.336 (Q^2 / 7000^2) = \cancel{6.26}\ 4.77 \times 10^{-8}\ Q^2$

VERMONT YANKEE DESIGN ENGINEERING
CALCULATION NO. VYC- 808
REVISION NO. 5
ATTACHMENT 7 PAGE 12 OF 12

YANKEE ATOMIC ELECTRIC COMPANY CALCULATION NO. ~~VYC-1389~~ PAGE 15 OF

SUBJECT Residual Heat Removal (LPCI) NPSH Evaluation

PREPARED BY ______ DATE ______ REVIEWED BY ______ DATE ______ WORK ORDER NO. 4922

Case II Friction Loss :

$H_f = H_f$ common + H_f single

and $H_{fa} = H_{fb} \, (Q_a^2 / Q_b^2)$ [Ref.(c)]

as a function of flow where $Q_{common} = 2\, Q_{single}$ for Case II.

$H_{fII} = H_{fc0} \, (Q^2 / Q_0^2) + H_{fs0} \, (Q^2 / Q_0^2)$

where H_{fc0} = friction loss at Q_0 for the common piping

and H_{fs0} = friction loss at Q_0 for the single piping

For Q_0 = 7000 gpm (one pump flow), friction loss for 20" Sched. 40 pipe is 0.376 psi per 100' [Ref.(c)]

For Q_0 = 14000 gpm (two pump flow), friction loss for 20" Sched. 40 pipe is 1.43 psi per 100' [Ref.(c)]

$H_{fc0} = L_{COM} \, (H_{f0})$

$= (\cancel{145.79} \; 61.75) (1.43 / 100) (2.31 \text{ ft/psi}) = \cancel{4.816'} \; 2.040 \; \cancel{2.039}$

$H_{fs0} = L_{T-C} \, (H_{f0})$

$= (206.91) (0.376 / 100) (2.31 \text{ ft/psi}) = 1.80'$

$H_{f0II} = H_{fc0} + H_{fs0} = \cancel{4.816} \; 2.04 + 1.80 = \cancel{6.616'} \; 3.84'$ @ 7000gpm per pump

therefore,

$H_{fII} = \cancel{4.816} \; 2.04 \, (Q_C^2 / 14000^2) + 1.80 \, (Q_S^2 / 7000^2)$

and for $Q_C = 2Q_S$

$H_{fII} = \cancel{4.816} \; 2.04 \, ((2Q_S)^2 / 14000^2) + 1.80 \, (Q_S^2 / 7000^2)$

$= \cancel{9.828\times10^{-8}} \; 4.163\times10^{-8} \, Q^2 + 3.673\times10^{-8} \, Q^2 = \cancel{1.350\times10^{-7}} \; 7.836\times10^{-8} \, Q^2$

Attachment 8

VY CALCULATION REVIEW FORM

Calculation Number: VYC-808 Revision Number: 6 CCN Number: NA

Title: Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident

Reviewer Assigned: W. Timofeev Required Date: NA

☐ Interdiscipline Review ☒ Independent Review

Comments*	**Resolution**
1. Page 7 – Need page numbers for all sections	✓1. OK
2. Attachment 6 page 3 – add "ft" to 137	✓2. OK
3. Page 17 – Add "F" to "riction"	✓3. OK
4. Page 18 bottom of page – add "w" to "here"	✓4. OK
5. Page 3 & 25 Ref. 3 – Calculation 1677 pending approval	✓5. Calculation approved.
6. Page 13 & 14 Sect. 2.2, 2.3 & 2.4 should be renumbered	✓6. OK
7. Page 10 & 11 for Vendor recommendation add Ref. Attach 5	✓7. OK
8. Conclusion should address RHR & Core Spray fibrous margin for debris plugging on the Intake Strainers	8. Revised

[signature] / 12-12-00
Reviewer Signature / Date

BC Slifer / 12/13/00
Calculation Preparer (Comments Resolved) / Date

Method of Review: ☒ Calculation/Analysis Review
☐ Alternative Calculation
☐ Qualification Testing

[signature] / 12-13-00
Reviewer Signature (Comments Resolved) / Date

*Comments shall be specific, not general. Do not list questions or suggestions unless suggesting wording to ensure the correct interpretation of issues. Questions should be asked of the preparer directly.

The limiting postulated event for drywell-to-wetwell bypass leakage is not the DBA-LOCA. Therefore changes in the DBA-LOCA pressure or temperature response do not impact bypass leakage requirements. The maximum bypass leakage will occur for a break size that maintains a drywell-to-suppression chamber pressure difference that is just less than that required to clear the drywell vents and for the longest credible duration. For this limiting break size, i.e., the break size with the minimum associated allowable leakage area, sufficient break flow is injected into the drywell to maintain a steady pressure difference between the drywell and suppression chamber while not clearing the drywell vent. [[

]]

Since the primary factors ([[ ]]) affecting the peak containment pressure during steam bypass events are not adversely impacted by power uprate, the existing criteria for drywell bypass leakage for VYNPS remain applicable at the uprated power conditions.

Therefore, since steam bypass for the limiting condition is independent of reactor power level, it is not adversely impacted by power uprate and the existing criteria for drywell bypass leakage for VYNPS remain applicable at uprated power conditions.

RAI SPSB-C-25

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, the Hydraulic Institute recommends margin above the required NPSH to suppress cavitation. What margin is needed for the VYNPS pumps crediting containment accident pressure and how is this margin accounted for in the VYNPS NPSH calculations? Provide quantitative information.

Response to RAI SPSB-C-25

The required NPSH ($NPSH_R$) information provided for the Vermont Yankee Nuclear Power Station (VYNPS) core spray (CS) and residual heat removal (RHR) pumps by the manufacturer specifically address time-phased operational requirements with low available NPSH ($NPSH_A$). No specific margin is included or required in the $NPSH_A$ calculation. However, there is some margin between the overpressure required and the credited overpressure requested and more margin to the overpressure available.

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2 Pages

The following general discussion provides additional background information regarding the topic of NPSH margin for pumps:

> The two primary bases for requiring levels of $NPSH_A$ above $NPSH_R$ are hydraulic performance and mechanical reliability. By meeting or exceeding the $NPSH_R$ for a particular flow or range of flows, hydraulic performance is maintained and mechanical reliability is assured for extended operation.
>
> Hydraulic performance can be reduced below the non-cavitating performance curve with reduced margins of NPSH. This degradation is typically less than margins provided for in the sizing of a pump to deliver its design performance.
>
> For a given pump design, the mechanical impact to impeller surfaces and other parts of the pump due to cavitation is determined by the frequency of such operation, the duration and the severity of the event(s), as well as material durability. Typically, all pumps are exposed to brief periods of cavitation during startup or other major system upsets with little, if any, measurable impact.
>
> Pumps installed in safety systems are fitted with materials of construction and mechanical parts that are qualified for extensive operating periods and frequent cyclic operation well beyond their expected service life.
>
> Although certain safety-related pumps can be described as having moderate suction energy levels, the frequency and duration of the events when $NPSH_A$ levels are at or near defined $NPSH_R$ levels, are minor when compared to the long-term design qualification of the pump.

RAI SPSB-C-26

With respect to the application dated September 10, 2003 (Reference 1), Attachment 6, Section 4.2.6, please provide for NRC review the VYNPS calculations of NPSH and containment accident pressure associated with the EPU amendment request.

Response to RAI SPSB-C-26

The NPSH calculations are documented in VYNPS calculations VYC-0808, Rev. 6, CCN05 and VYC-2314, Rev. 0 and are included in Attachment 4 of this submittal as Exhibits 1 and 2, respectively. Based on discussions with NRC staff, it is understood that providing these calculations should be sufficient for NRC staff review needs.

The calculation of containment accident pressure, used as input to the LOCA NPSH calculation, was performed by GE.

GE Nuclear Energy

175 Curtner Ave., San Jose, CA 95125

NEDO-33090
Revision 0
Class III
0000-0007-5271
September 2003

SAFETY ANALYSIS REPORT

FOR

VERMONT YANKEE NUCLEAR POWER STATION

CONSTANT PRESSURE POWER UPRATE

Prepared by: E. D. Schrull

Approved by:
Michael J. Dick, Project Manager
General Electric Company

Approved by:
Craig J. Nichols, Project Manager
Entergy Nuclear Operations, Inc.

Docket No. 50-271
DPS Exhibit #18
7 Pages

The higher suppression pool temperature (194.7°F) and containment pressure during a postulated LOCA (Section 4.1) do not affect hardware capabilities of CS equipment, except for the CS pump seals.

The peak suppression pool temperature during a limiting LOCA remains below the CS pump seal design temperature of 210°F. However, this temperature exceeds the maximum operating temperature of 185°F analyzed for the pump seals. Either the pump seals will be re-qualified for the peak suppression pool temperature, or a modification will be completed to ensure seal operation prior to the CPPU implementation.

[[

]]

4.2.4 Low Pressure Coolant Injection

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to help maintain reactor vessel coolant inventory for a large break LOCA and for any small break LOCA after the reactor vessel has depressurized. The LPCI operating requirements are not affected by CPPU. The adequacy of this system is demonstrated by the margins discussed in Section 4.3.

[[

]]

4.2.5 Automatic Depressurization System

The ADS uses SRVs to reduce the reactor pressure following a small break LOCA when it is assumed that the high-pressure systems have failed. This allows the CS and LPCI to inject coolant into the reactor vessel. The adequacy of this system is demonstrated by the margins discussed in Section 4.3. [[

]]

4.2.6 ECCS Net Positive Suction Head

Following a LOCA, the RHR and CS pumps operate to provide the required core and containment cooling. Adequate NPSH is required during this period to assure the essential pump

operation. The NPSH for the ECCS pumps was evaluated for the limiting conditions following a DBA LOCA. The limiting NPSH conditions occur during long-term post-LOCA pump operation and depend on the total pump flow rates, debris loading on the suction strainers, and suppression pool temperature.

The NPSH for each pump was calculated based on the expected flow rates during the short-term and long-term ECCS pump operation. The pump flow rates for the short-term case are 7400 (14,200) gpm total RHR flow for single (two) pump operation and 4600 gpm total CS flow. The pump flow rates for the long-term case are 7400 gpm total RHR flow and 3500 gpm total CS flow. The debris loading on the suction strainers and the methodology used to calculate available ECCS NPSH for CPPU are the same as the pre-CPPU conditions. The containment response temperature and pressure profiles are CPPU specific. CPPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following a LOCA. As a result, the suppression pool water temperature and containment pressure increase.

The assumptions used in the CPPU containment response analyses maximize the suppression pool temperature and minimize the containment pressure. These include operation of the RHR pumps for containment cooling in the containment spray mode after 10 minutes. The analyses then assume that the operators establish long-term containment cooling and control ECCS flow.

Short-term and long-term containment analyses were performed for the CPPU conditions (short-term from 0 to 600 seconds and long-term from 0 until the end of the event). The short-term containment analysis shows that the peak suppression pool temperature of 165.1°F occurs at 600 seconds after the LOCA event when the suppression pool pressure is 17.75 psia. The long-term containment analysis shows that the peak suppression pool temperature of 194.7°F occurs at 24,094 seconds after the LOCA event when the suppression pool pressure is 22.77 psia. The NPSH analyses conclude that containment overpressure is needed to meet long-term NPSH requirements. Table 4-2 shows the overpressure credit and Figure 4-6 shows the containment overpressure available, the required overpressure, the overpressure credit, and NPSH margins during the long-term DBA LOCA at CPPU conditions.

Based on the above, VYNPS is requesting approval of the "stepped" overpressure credit shown in Table 4-2 and Figure 4-6 to meet DBA LOCA long-term NPSH requirements.

RHR is required to operate during an ATWS event. CPPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following this event (see Section 9.3.1). As a result, the peak suppression pool water temperature and peak containment pressure increase. The NPSH evaluation at these peak pool temperatures shows that adequate overpressure is available to satisfy NPSH requirements for these pumps during an ATWS event.

RHR is also required to operate during SBO and Appendix R fire events. CS is required to operate during an Appendix R fire event. CPPU RTP operation increases the reactor decay heat, which increases the heat addition to the suppression pool following these events (see Sections 6.7.1 and 9.3.2). As a result, the peak suppression pool water temperature and peak

containment pressure increase. The NPSH evaluation at these peak pool temperatures shows that adequate overpressure is available to satisfy NPSH requirements for these pumps.

The HPCI system primary function is to provide reactor inventory makeup water and assist in depressurizing the reactor during an intermediate or small break LOCA. HPCI system operation is also credited during ATWS, Appendix R, and SBO events. The available NPSH and required NPSH for the HPCI pump are not changed for CPPU, because the system configuration and the specified operational temperature limit for the process water do not change.

4.3 EMERGENCY CORE COOLING SYSTEM PERFORMANCE

The VYNPS ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance characteristics will not be changed for CPPU. ECCS-LOCA performance analyses demonstrate that the 10 CFR 50.46 requirements continue to be met at the CPPU RTP conditions. The VYNPS topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Large break peak clad temperature – limiting case	[[	
Large break peak clad temperature - limiting event analysis		
Small break peak clad temperature – break spectrum		
Small break peak clad temperature – ADS capacity		
Local cladding oxidation		
Core wide metal water reaction		
Coolable geometry		
Long-term cooling		]]

[[

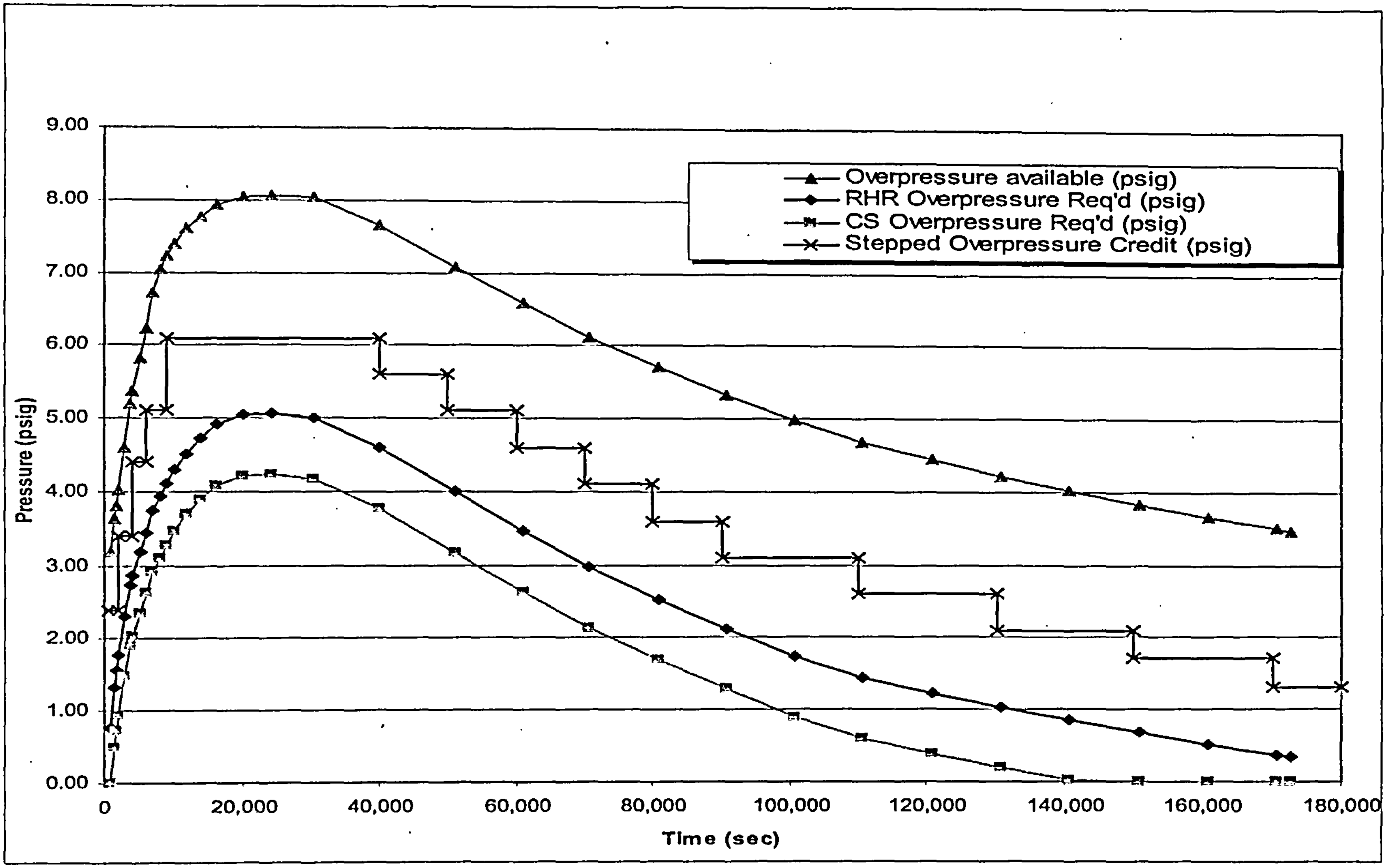
]] The break spectrum response is determined by the ECCS network design and is common to all BWRs. [[

Table 4-2
VYNPS Overpressure Credit for NPSH DBA LOCA – Long-Term

Time After LOCA (sec)	Overpressure Credit (psig)
601	2.4
2,000	2.4
2,001	3.4
4,000	3.4
4,001	4.4
6,000	4.4
6,001	5.1
9,000	5.1
9,001	6.1
40,000	6.1
40,001	5.6
50,000	5.6
50,001	5.1
60,000	5.1
60,001	4.6
70,000	4.6
70,001	4.1
80,000	4.1
80,001	3.6
90,000	3.6
90,001	3.1
110,000	3.1
110,001	2.6
130,000	2.6
130,001	2.1
150,000	2.1
150,001	1.7

Time After LOCA (sec)	Overpressure Credit (psig)
170,000	1.7
170,001	1.3
180,000	1.3

Figure 4-6
Overpressure Required for NPSH DBA LOCA – Long-Term



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STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE

June 8, 2004

RE: Vermont Yankee Nuclear Power Station
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 263
Extended Power Uprate - State of Vermont Comments

Richard Ennis, Project Manager
U.S. Nuclear Regulatory Commission
Washington, D.C., 20555

Dear Mr. Ennis,

The state of Vermont, through its NRC state liaison officer, makes the requests identified below of the Nuclear Regulatory Commission staff (NRC) with regard to its review of the proposed Vermont Yankee power uprate. Vermont asks that NRC perform independent calculations in three areas to confirm the adequacy of the proposed uprate: 1) the adequacy the steam dryer with power uprate flow rates, 2) credit for containment overpressure for net positive suction head (NPSH) adequacy, and 3) flow-induced vibration adequacy of the main steam and feedwater systems. This request is consistent with NRC's Review Standard for Extended Power Uprates (RS-001).

Background

On March 15, 2004, the Vermont Public Service Board requested the NRC perform an *independent engineering assessment*[1] of Vermont Yankee related to its proposed 20% power uprate. NRC responded on May 4, 2004, stating it would perform a new engineering assessment inspection at Vermont Yankee. In its May 4, 2004, letter, NRC also identified that its power uprate review consisted of a comprehensive assessment of engineering, design and safety analyses comprising about 4000 staff-hours.

[1] The PSB created the term, *independent engineering assessment*, which it defined within its March 15, 2004 request as a level of effort of four persons for four weeks.

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DPS Exhibit #19
5 Pages

Also, in December 2003, the NRC issued Revision 0 of RS-001. In response to comments from the Advisory Committee on Reactor Safeguards (ACRS), NRC included the following statement regarding independent calculations:

> *Perform audits and/or independent calculations as deemed necessary and appropriate to support review of the licensee's application. In determining the need for performing audits and/or independent calculations, consider the following:*
>
> - *confidence of the NRC staff in the models and/or methods used by the licensee*
> - *confidence of the NRC staff in the analysis results*
> - *familiarity of the NRC staff with the models and/or methods used by the licensee*
> - *prior use of the models and/or methods for similar plant designs and operating conditions and the NRC staff's experience related to such use*
> - *NRC staff experience with the impact of proposed changes on analysis results*
> - *available margin versus level of uncertainty in analysis results*
> - *efficiency gains that may result from performing audits and/or independent calculations*
>
> RS-001, Section 2.1, page 2.1-3.

Accordingly, we believe that independent calculations should be performed by NRC as part of the new engineering assessment inspection, together with the power uprate review, in the three areas identified below.

<u>Steam Dryer Analysis</u>

Despite licensee and industry analysis, significant, power uprate related failures of steam dryers have occurred at four units - Quad Cities 1 & 2 and Dresden 2 & 3.Of three types of steam dryers, square, curved and slanted, Vermont Yankee has the same squared-design steam dryer as Quad Cities and Dresden, determined to be the most susceptible to power uprate related cracking.

In NRC's letter of May 4, 2004, it was stated that outside technical experts are assisting NRC staff on steam dryer issues. In addition, we are aware that Entergy has performed an analysis of its steam dryer and has completed modifications for power uprate in its Spring 2004 refueling outage. In addition, Entergy discovered and dispositioned numerous cracks in the steam dryer.

We believe the analysis for the adequacy of the steam dryer meets the criteria for independent calculation stated in RS-001, Section 2.1. Therefore, we request that NRC verify by independent calculation the adequacy of Vermont Yankee's steam dryer, with modifications, for power uprate as part of its new engineering assessment inspection, together with the power uprate review. Further, we request that Vermont Yankee not be allowed to operate above original licensed thermal power (OLTP) until the NRC verification analysis of the steam dryer is completed.

Credit for Containment Overpressure

Centrifugal pumps required to perform safety actions must have adequate NPSH in order to function properly. For power uprate situations, available NPSH is reduced because water temperatures are warmer than at original power because more heat is produced in the reactor. To compensate for decreased NPSH because of hotter water temperatures, Entergy requests credit for the elevated pressure in containment (containment overpressure). In Section 4.2.6 of the *Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate (PUSAR)*, NEDC-33090, September 2003, Entergy requests containment overpressure credit for either one or two sets of pumps for four different situations:

- On loss of coolant accidents (LOCAs), for the residual heat removal (RHR) and core spray (CS) pumps
- On an anticipated transient without scram (ATWS), for the RHR pumps
- On station black outs (SBOs), for the RHR pumps
- On Appendix R fire events, for the RHR and CS pumps

In our letter of December 8, 2003[2], we asked NRC questions about granting containment overpressure credit, which represents both a change in Vermont Yankee's design basis and a change in NRC's regulatory policy. It does not appear that granting containment overpressure credit is *necessary* in the context of Draft Regulatory Guide DG 1107, at 7, and it appears that the design can be *practicably altered* in the context of DG 1107, at 16, by operation at OLTP. Therefore, pending response to our December 8, 2003 letter, we do not believe containment overpressure credit should be allowed.

Notwithstanding, and without waiving our belief that containment overpressure credit should not be allowed, if such credit is allowed, we believe the NRC should perform the following independent calculations.

---

[2] We are awaiting response to our letter of December 8, 2003.

The four situations for which containment overpressure credit is requested are fundamentally different. Two situations, LOCA and ATWS pressurize the drywell first and then the torus. The other situations, SBO and Appendix R events, pressurize only the torus. The analysis of each situation consists of a containment response analysis and an NPSH calculation. Finally, the single failure criteria effects are not the same for each situation.

Because of the importance of the RHR and CS pumps for the situations in question, and because of the controversial nature of the change in NRC's regulatory policy, we believe these situations meet the requirements of RS-001, Section 2.1 for independent calculations. Therefore, we request that NRC verify by independent calculation the adequacy of the claimed containment overpressure credit for power uprate as part of its new engineering assessment inspection, together with the power uprate review. The containment response for each situation where credit is requested should be independently verified by NRC analysis. A single failure mode and effects analysis should be performed by NRC for each situation and sufficient calculations should be performed to assure the most limiting single failure is identified[3]. The water temperature and available NPSH should be determined for each situation, again assuming the most limiting single failure, to verify the calculated containment overpressure provides sufficient NPSH.

Flow-Induced Vibration Adequacy

In *PUSAR* Section 3.4.1, it is stated that Entergy will demonstrate the adequacy of increased flow-induced vibration of the main steam system and feedwater system piping only through a piping startup testing program. However, since power uprate related, vibration failures have occurred for an electromatic releif valve, small piping in main steam and feedwater lines, and a feedwater instrument probe, we believe the flow-induced adequacy of the main steam and feedwater lines, including branch lines connected to the main steam and feedwater systems, should be confirmed by analysis wherever possible.

---

[3] With regard to the single failure mode and effects analysis, we believe the guidance from Regulatory Guide 1.183, Section C.5.1.4, albeit for a different subject - *alternative source term*, is sound and should be applied for the review of *containment overpressure credit*. In summary, Section C.5.1.4 states that, since a request for *alternative source term* is a change to a plant's historical licensing basis, the review of its adequacy may consider current, rather than historical, licensing requirements for other affected aspects of the request. Since *containment overpressure credit* is a change to Vermont Yankee's historical licensing basis, its adequacy should be evaluated using the single failure criteria applicable to current-day license evaluations.

Since failures have occurred in this area, we believe the area of flow-induced vibrations meet the requirement of RS-001, Section 2.1 for independent calculations. Therefore we request that NRC verify by independent calculation the adequacy of increased flow-induced vibration of the main steam and feedwater systems, including branch lines, as part of its new engineering assessment inspection, together with the power uprate review.

Conclusion

RS-001, Section 2.1 identifies either audits or independent calculations as appropriate actions for the conditions identified on page 2.1-3. We believe that independent calculations by the NRC should be performed for the three areas identified above. However, we would be pleased to discuss with the NRC whether audits of any of these areas is more appropriate than independent calculations. We welcome the opportunity to provide these comments and look forward to resolving these issues in a satisfactory manner. If you have questions about these items, please call me at 802-828-2321, or Mr. William Sherman of my staff at 802-828-3349.

Sincerely,

David O'Brien, Commissioner
State Liaison Officer

cc: Mario V. Bonaca, Chairman, ACRS
J. Thayer, Entergy
Sen. Patrick Leahy
Sen. James Jeffords
Rep. Bernard Sanders

MINOR CALCULATION CHANGE FORM (Tracked via DRN in MERLIN)

Calculation No.: VYC-0808	**Revision:** 6
Calculation Title: Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident or Anticipated Transient Without Scram	**Indicate Status of Minor Calculation Change:** ~~Prel:~~ Pend: ~~As-Built~~
MERLIN DRN No. or Minor Calculation Change No.: 05	
Modification No./Task No./ER No. VYDC-2003-008	
Computer Code Used ☐ Yes ☒ No. **If "Yes", Code:** ________	

1 Purpose of Change:	Incorporate GE-VYNPS-AEP-346
2 SSC affected:	See attached VYAPF0017.07
3 Design Input Documents not used in parent Calculation:	See attached VYAPF0017.07
4 Drawings/Procedures/ Calculations / other Documents affected	See attached VYAPF0017.07
5 Description of Change:	See attached description
6 Impact on existing calculation conclusion:	Revised containment overpressure envelope for long term LOCA
7 Impact on DBD's, UFSAR, Technical Specifications:	See attached VYAPF0017.07

8. The existing calculation does/does not (circle one) have a calculation verification checklist. (See Remarks)

Remarks: This ENN-DC-126 MCC is to a VY design verified calculation prepared under AP-0017. AP-0017 did not have a "checklist", instead design verification was documented on form VYAPF0017.04

NOTE:
A. If UFSAR or Technical Specifications need to be revised, Minor Calculation Change Form should not be used unless it is an editorial change to the UFSAR or Technical Specifications.
B. Minor Calculation Change Forms do not change the status of the Parent Calculation Revision.

	Print/Sign		
Prepared by:	E. P. O'Brien EPOBr	Date:	7/1/04
Reviewed by: *	E.G. Lind E.G. Lind	Date:	7/1/04
Approved by	D.E. Yasi D E Yasi	Date:	7/1/04

* Where the original calculation was design verified, the reviewer signature confirms the latest design verification is still valid.

This IS a Quality Record -

Docket No. 50-271
DPS Exhibit #20
11 Pages

5. Description of Change

The NPSH evaluation of the RHR and CS pumps is performed, using the same methodology as CCN04, for long term DBA-Loss of Coolant Accident (LOCA) case only. This CCN uses the EPU suppression pool temperature/pressure data supplied in Reference 1. Note – all References contained in AP0017.07 (Attachment B)

Reason for Change:
This CCN provides additional results regarding the Residual Heat Removal (RHR) and Core Spray (CS) Pump NPSH resulting from the Extended Power Uprate (EPU), which was initially evaluated in CCN04.
Specifically, this CCN05 updates the post LOCA results based on revised containment temperature and pressure profiles provided in Reference 1. The revised containment profiles reflect two changes from the CCN04 EPU evaluation: (1) The post LOCA time was extended to 200,000 seconds, beyond which overpressure is no longer being required, and (2) the containment leakage was increased from 0.8 to 1.5 weight percent per day, to reflect the recently submitted Alternate Source Term (AST) License Amendment Request (Ref: 5, ERC-2004-024)

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Note: No Table 4.1 or Figure 4.1 are contained in this CCN.

TOTAL PAGES: 17

1.0 Objective

The objective of this CCN is to update the CCN04 post LOCA Extended Power Uprate (EPU) evaluation of the adequacy of the available NPSH for the Residual Heat Removal (RHR) and Core Spray (CS) pumps. This includes identification of any change in the amount and duration of suppression pool (torus) overpressure required to maintain adequate NPSH.

Specifically, this CCN05 updates the post LOCA results based on revised containment temperature and pressure profiles provided in Reference 1. The revised containment profiles reflect two changes from the CCN04 EPU evaluation: (1) The post LOCA time was extended to 200,000 seconds, beyond which overpressure is no longer being required, and (2) the containment leakage was increased from 0.8 to 1.5 weight percent per day, to reflect the recently submitted Alternate Source Term (AST) License Amendment Request (Ref: 5, ERC-2004-024)

2.0 Methodology

General

The methodology for determining the NPSH available (NPSHa) for a given event and temperature is the same as that developed in VYC-0808 Rev 6 and presented in Table 1 of that calculation (Ref: 4). The NPSH required (NPSHr) is also per VYC-0808 Rev 6 and is discussed in detail in Section 4.0 of this CCN. The methodology for determining the pump suction strainer head loss during a LOCA, and the time dependent profile for required overpressure is the same as that developed in CCN04.

3.0 Assumptions

1. None made

4.0 Analysis

As stated in Section 2.0, the methodology for determining the NPSHa is the same as that developed in VYC-0808 Rev 6 and presented in Table 1 of that calculation (Ref: 4). The following terms are used in the evaluation.

NPSHa (ft) = net positive suction head available without overpressure credit
(14.7-Pg)(144vf)+Z-hf-hs-hd

where:

Z (ft) = suction elevation head

hf (ft) = suction line losses

hs (ft) = clean strainer losses

hd (ft) = strainer debris losses

Pg (psia) = vapor pressure @ torus temperature

vf (ft^3/lb) = specific volume @ torus temperature and pressure

Pg and vf are obtained from ASME Steam Tables 1967 Formulation (Ref: 7)

NPSHr (ft) = net positive suction head required.

It should be noted that the NPSH required data provided by the pump vendor, as documented in Figures 2.1-1 and 2.2-1 of Attachment 3 to calculation VYC-0808, is actually *Allowable Operating Periods @ NPSHa Specified.* Allowable hours of operation at specified NPSHa values are identified for a range of flows. For this CCN, the NPSHa specified in these Figures is taken as the NPSHr at a given operating time.

Q (gpm) = pump flow rate

OPR (psig) = Overpressure Required
(NPSHr – NPSHa)/(144*Vf)

For those profile points where there is inadequate NPSH, when considering the suppression pool pressure to be atmospheric (14.7 psia), OPR is the amount of suppression pool pressure required to make NPSHa (ft) equal to NPSHr (ft).

OPA (psig)– Overpressure Available
The suppression pool pressure available, above atmospheric, for a given event and time.

OPC (psig)– Overpressure Credit Taken
The overpressure credited in the evaluation of NPSH. Engineering judgement is used to select the credit to be greater than the OPR, by a reasonable amount, and less than the OPA.

Detailed discussion of the above terms is provided in the subsections that follow.

4.1 LOCA – Long Term

The temperature and pressure (T/P) profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-346 (Ref: 1). The long term data is provided from 0-864,000 seconds.

The evaluation of NPSH is documented in Table 4.2 using selected T/P points representing the long term profile of the suppression pool. The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for the T/P profile of CS and RHR. The evaluated long term flow rates of 7400 gpm (RHR) and 3500 gpm (CS) are consistent with calculation VYC-0808 Rev 6 (Ref: 4). Further discussion of selected terms is presented below.

Suction Elevation Head, Z

The values of Z for RHR and CS (12.40' and 12.57' respectively) as calculated in Section 3.5 of VYC-0808 are conservatively used in the evaluation. The suction elevation head is based on the water elevation in the torus. The EPU suppression pool water volume is slightly larger than the existing value used in VYC-0808, which would result in a slight increase in water elevation, and therefore Z.

A water volume comparison at maximum pool temperature is provided below:

	Pre-EPU	EPU
Ref:	(VYC-0808 Rev 6 Section 3.5)	(GE-VYNPS-AEP-346)
Long term	77,640 cuft	79,470 cuft

Maximum Debris Losses (hd)

1 RHR: CCN #3 (Ref: 3) calculated the limiting head loss as 0.24 ft at 181.7°F and 7400 gpm. Note that this is a slight reduction from the head loss (0.33 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.33 ft at 173°F is used. (Case 1 of Tables 2 and 8 of Ref: 2).

CS: Note that CCN #3 (Ref: 3) documents the up-to-date limiting head loss as 0.19 ft at 181.7°F and 3500 gpm. This is a slight reduction from the head loss (0.21 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.21 ft at 173°F is used. This is based on a conservative CS flow rate of 4000 gpm. (Case 3b of Tables 2 and 8 of Ref: 2).

NPRHr - CS

Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 3500 gpm the allowable NPSH increases between 7 and 20 hrs of operation and a value of 29.6 ft is acceptable beyond 20 hrs of operation. This maximum value is conservatively used for the entire long term period (>600 sec).

NPRHr - RHR

Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 7400 gpm the allowable NPSH increases between 7 and 100 hrs of operation and a value of 31.7 ft is acceptable beyond 100 hrs of operation. This maximum value is conservatively used for the entire long term period (>600 sec).

4.1.1 Evaluation

As can been seen from Figure 4.2 the overpressure required for RHR envelopes that required for CS and the overpressure varies continuously over time. In order to facilitate reporting and presentation of the overpressure required, an enveloping, stepped, overpressure credit is overlaid on Figure 4.2. Refer to Section 4.0 for discussion on selection of overpressure credit.

Though the long term flow rates are postulated at time 600 seconds (e.g. CS throttled down from 4600gpm to 3500gpm), it is not the intent of this calculation to imply at what time throttling should commence or how much throttling is required. This is a function of the time dependent NPSHr and pool temperature. This calculation conservatively evaluates the maximum NPSHr as occurring over the entire operating period (>600 sec). The actual NPSHr is lower between 0-7 hrs and increases after 7 hrs.

5.0 Summary of Results

NPSHa is rounded to the nearest 0.1ft and OPR, OPC, and OPA are rounded to the nearest 0.1psig.

5.1 LOCA - Long Term (>600 sec):

NPSHa is adequate for both CS and RHR pumps with an overpressure credit that varies over time, as shown in Fig. 4.2. NPSHa, OPR, OPC, OPA are shown below, at the peak temperature

Pump	Total flow, gpm	NPSHr, ft	NPSHa, ft	OPR, psig	OPC, psig	OPA, psig
CS	3,500	29.6	19.5	4.2	6.1	8.0
1RHR	7,400	31.7	19.6	5.1	6.1	8.0

6.0 Conclusions

Torus overpressure must be credited for 200,000 seconds for operating the RHR and CS pumps at EPU conditions for a long term DBA-Loss of Coolant Accident (LOCA) to achieve adequate NPSH available.

The results of this CCN will provide input to the PUSAR (Ref: 12) for the RHR and CS NPSH evaluation and will alter input to calculation VYC-1628 (Ref: 13) to address the need for crediting torus overpressure in the calculation of NPSH available. Note that calculation VYC-1628 may be superseded by GE EPU Analysis. The need for crediting torus overpressure in the RHR and CS NPSH evaluation, shall also be addressed in the SADBD (Ref: 14), UFSAR (Ref: 15), and system DBDs RHR (Ref: 16) and CS (Ref: 17).

Note that the changes to the UFSAR were originally proposed in CCN04 and are pending incorporation via the design change and licensing processes. This MCC simply updates the previous CCN04 performed under AP-0017. The UFSAR does not currently contain information on containment overpressure.

Note that use of overpressure credit must be approved by the NRC as part of EPU.

No specific 50.59 Screening/Evaluation is required for this CCN since all EPU design changes and associated 50.59 documentation will be part of VYDC-2003-008.

Table 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage)

LOCA - Long

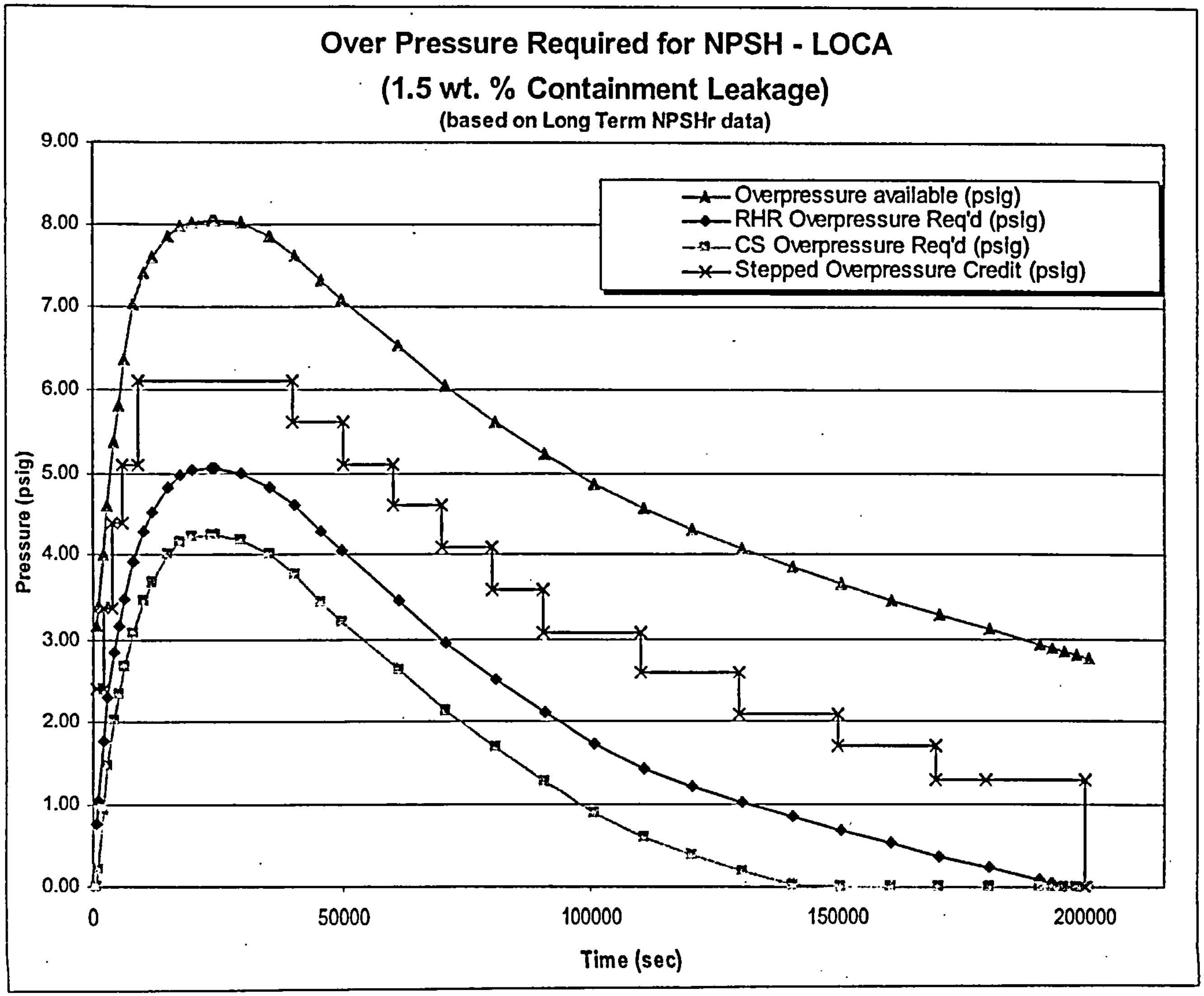
NPSHa =	(14.7-Pg)(144Vf)+Z-hf-hs-hd			Cross references: Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
OPR =	Over pressure required (NPSHr – NPSHa)/(144*Vf)			See Discussion in Section 4.0 of this CCN
OPA =	Over pressure available			
OPC =	Over pressure credited			
Long Term Flow Rate (gpm)				
1 RHR	Q = 7400	CS	Q = 3500	Table of 1 calc VYC-0808 Rev 6 (Ref: 4)
Suction Line Losses (ft)				
1 RHR	hf = 4.77E-8*Q^2	CS	hf = 2.5E-7*Q^2	Section 3.7 of VYC-0808 Rev 6 (Ref: 4)
Clean Strainer Losses (ft)				
1 RHR	hs = 0.33	CS	hs = .38*(Q/4000)^2 for Q<= 4000	Section 3.6 of VYC-0808 Rev 6 (Ref: 4)
Maximum Debris Losses (ft) @ >= 173F				
1 RHR	hd = 0.33	CS	hd = 0.21	See discussion in Section 4.1 of this CCN
Maximum Debris Losses (ft) @ < 173F				
1 RHR	hd = .33*(173/T)	CS	hd = .21*(173/T)	See discussion in Section 2.0 of VYC-0808 Rev 6 CCN 4 (Ref: 11)
where T = suppression pool temperature, F				
Elevation Head (ft)				
RHR	Z = 12.4	CS	Z = 12.57	See discussion in Section 4.1 of this CCN for conservatism.
NPSHr (ft)				
1 RHR	NPSHr = 31.7	CS	NPSHr = 29.6	See discussion in Section 4.1 of this CCN

Table 4.2 - LOCA – Long Term (1.5 wt. % Containment Leakage)

RHR - Long Term (After EPU) 1.5 wt. % Containment Leakage

Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	RHR NPSHa (ft)	RHR NPSHr (ft)	RHR OPR (psig)	OPA (psig)	OPC (psig)
778.85	169.6	17.88	5.938	0.016448	12.40	2.61	0.33	0.34	29.87	31.70	0.77	3.18	2.40
1088.64	171.7	18.12	6.231	0.016460	12.40	2.61	0.33	0.33	29.20	31.70	1.05	3.42	2.40
2022.07	176.6	18.72	6.962	0.016489	12.40	2.61	0.33	0.33	27.50	31.70	1.77	4.02	3.40
2950.64	180	19.3	7.511	0.016509	12.40	2.61	0.33	0.33	26.22	31.70	2.31	4.60	3.40
4185.2	183.3	20.07	8.078	0.016529	12.40	2.61	0.33	0.33	24.89	31.70	2.86	5.37	4.40
5108.82	185.1	20.5	8.402	0.016541	12.40	2.61	0.33	0.33	24.13	31.70	3.18	5.80	4.40
6241.48	186.9	21.05	8.737	0.016552	12.40	2.61	0.33	0.33	23.34	31.70	3.51	6.35	5.10
8035.1	189.1	21.72	9.161	0.016566	12.40	2.61	0.33	0.33	22.34	31.70	3.92	7.02	5.10
10190.79	191	22.09	9.541	0.016578	12.40	2.61	0.33	0.33	21.44	31.70	4.30	7.39	6.10
12039.95	192.1	22.29	9.767	0.016585	12.40	2.61	0.33	0.33	20.91	31.70	4.52	7.59	6.10
15125.17	193.6	22.55	10.083	0.016594	12.40	2.61	0.33	0.33	20.16	31.70	4.83	7.85	6.10
17625.17	194.3	22.67	10.233	0.016599	12.40	2.61	0.33	0.33	19.81	31.70	4.98	7.97	6.10
20063.67	194.6	22.71	10.298	0.016601	12.40	2.61	0.33	0.33	19.65	31.70	5.04	8.01	6.10
23735.2	194.7	22.74	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	8.04	6.10
24093.8	194.7	22.74	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	8.04	6.10
24359.79	194.7	22.74	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	8.04	6.10
29855.07	194.4	22.71	10.255	0.016599	12.40	2.61	0.33	0.33	19.75	31.70	5.00	8.01	6.10
35212.2	193.6	22.54	10.083	0.016594	12.40	2.61	0.33	0.33	20.16	31.70	4.83	7.84	6.10
40032.64	192.5	22.31	9.851	0.016587	12.40	2.61	0.33	0.33	20.71	31.70	4.60	7.61	5.60
45400.32	190.9	22.01	9.521	0.016577	12.40	2.61	0.33	0.33	21.49	31.70	4.28	7.31	5.60
49775.32	189.8	21.77	9.300	0.016570	12.40	2.61	0.33	0.33	22.01	31.70	4.06	7.07	5.60
61022.64	186.7	21.22	8.699	0.016550	12.40	2.61	0.33	0.33	23.43	31.70	3.47	6.52	4.60
70699.13	183.9	20.74	8.185	0.016533	12.40	2.61	0.33	0.33	24.64	31.70	2.97	6.04	4.10
80699.13	181.3	20.3	7.730	0.016517	12.40	2.61	0.33	0.33	25.71	31.70	2.52	5.60	3.60
90697.01	178.8	19.93	7.313	0.016502	12.40	2.61	0.33	0.33	26.68	31.70	2.11	5.23	3.10
100697.01	176.4	19.56	6.931	0.016488	12.40	2.61	0.33	0.33	27.57	31.70	1.74	4.86	3.10
110697.01	174.4	19.26	6.625	0.016476	12.40	2.61	0.33	0.33	28.29	31.70	1.44	4.56	2.60
120697.01	172.9	19	6.404	0.016467	12.40	2.61	0.33	0.33	28.80	31.70	1.22	4.30	2.60
130684.82	171.5	18.77	6.202	0.016459	12.40	2.61	0.33	0.33	29.27	31.70	1.03	4.07	2.10
140684.83	170.2	18.56	6.020	0.016452	12.40	2.61	0.33	0.34	29.68	31.70	0.85	3.86	2.10
150684.83	169	18.36	5.856	0.016445	12.40	2.61	0.33	0.34	30.06	31.70	0.69	3.66	1.70
160684.83	167.7	18.17	5.683	0.016437	12.40	2.61	0.33	0.34	30.46	31.70	0.52	3.47	1.70
170684.83	166.5	18.01	5.526	0.016431	12.40	2.61	0.33	0.34	30.82	31.70	0.37	3.31	1.30
180684.83	165.3	17.83	5.374	0.016424	12.40	2.61	0.33	0.35	31.16	31.70	0.23	3.13	1.30
190684.83	164.1	17.65	5.225	0.016417	12.40	2.61	0.33	0.35	31.51	31.70	0.08	2.95	1.30
193184.83	163.8	17.6	5.188	0.016415	12.40	2.61	0.33	0.35	31.59	31.70	0.05	2.90	1.30
195684.83	163.5	17.56	5.151	0.016414	12.40	2.61	0.33	0.35	31.68	31.70	0.01	2.86	1.30
198184.83	163.2	17.52	5.115	0.016412	12.40	2.61	0.33	0.35	31.76	31.70	0.00	2.82	1.30
200684.83	162.9	17.47	5.079	0.016410	12.40	2.61	0.33	0.35	31.84	31.70	0.00	2.77	0.00

Figure 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage)



OPC overpress credit	
(sec)	(psig)
600	2.4
2000	2.4
2001	3.4
4000	3.4
4001	4.4
6000	4.4
6001	5.1
9000	5.1
9001	6.1
40000	6.1
40001	5.6
50000	5.6
50001	5.1
60000	5.1
60001	4.6
70000	4.6
70001	4.1
80000	4.1
80001	3.6
90000	3.6
90001	3.1
110000	3.1
110001	2.6
130000	2.6
130001	2.1
150000	2.1
150001	1.7
170000	1.7
170001	1.3
180000	1.3
200000	1.3
200001	0

Design Output Documents - This calculation provides output to the following documents. (Refer to Appendix A, section 5)

* Reference #	** DOC #	REV #	Document Title (including Date, if applicable)	**** Affected Program	†††Critical Reference (✓)
12	NEDC-33090P	0	Safety Analysis Report For VYNPS Constant Pressure Power Uprate (PUSAR)		
13	VYC-1628	0	Torus Temperature and Pressure Response to Large Break LOCA and MSLB Accident Scenarios (may be superseded be GE EPU analysis)		
14	SADBD	2/IC2	Topical Design Basis Document for Safety Analysis		✓
15	UFSAR	17	Updated Final Safety Analysis Report		✓
16	RHR	1/IC16	Design Basis Document for Residual Heat Removal System		✓
17	CS	0/IC10	Design Basis Document for Core Spray System		✓

* Reference # - Assigned by preparer to identify the reference in the body of the calculation.

** Doc # - Identifying number on the document, if any (e.g., 5920-0264, G191172, VYC-1286)

*** Document Title - List the specific documentation in this column. "See attached list" is not acceptable. Design Input/Output Documents should identify the specific design input document used in the calculation or the specific document affected by the calculation and not simply reference the document (e.g., VYDC, MM) that the calculation was written to support. If a DBD is used as a general reference, include the most current interim change number after the title.

**** Affected Program - List the affected program or the program that reference is related to or part of.

† If "yes," attach a copy of "VY Calculation Data" marked-up to reflect deletion (See Section 3.1.8 for Revision and 5.2.3.18 for CCNs).

†† If the listed input is a calculation listed in the calculation database that is not a calculation of record (see definition), place a check mark in this space to indicate completion of the required significant difference review. (see Appendix A, section 4.1.4.4.3). Otherwise, enter "N/A."

††† If the reference is UFSAR, DBD or Reload (IASD or OPL), check Critical Reference column and check UFSAR, DBD or Reload, as appropriate, on this form (above).

MINOR CALCULATION CHANGE FORM (Tracked via DRN in MERLIN)

Calculation No.: VYC-0808	**Revision:** 6
Calculation Title: Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident or Anticipated Transient Without Scram	**Indicate Status of Minor Calculation Change:** – Prel; – Pend; – As-Built
MERLIN DRN No. or Minor Calculation Change No.: 06	
Modification No./Task No./ER No. VYDC-2003-008 .	
Computer Code Used ☐ Yes ☒ No. If "Yes", Code:________	

1 Purpose of Change:	Incorporate GE-VYNPS-AEP-346R1
2 SSC affected:	See attached VYAPF0017.07
3 Design Input Documents not used in parent Calculation:	See attached VYAPF0017.07
4 Drawings/Procedures/ Calculations / other Documents affected	See attached VYAPF0017.07
5 Description of Change:	See attached description
6 Impact on existing calculation conclusion:	Revised containment overpressure envelope for long term LOCA using 100% Spray Efficiency
7 Impact on DBD's, UFSAR, Technical Specifications:	See attached VYAPF0017.07

8. The existing calculation does/does not (circle one) have a calculation verification checklist. (See Remarks)

Remarks: This ENN-DC-126 MCC is to a VY design verified calculation prepared under AP-0017. AP-0017 did not have a "checklist", instead design verification was documented on form VYAPF0017.04

NOTE:
A. If UFSAR or Technical Specifications need to be revised, Minor Calculation Change Form should not be used unless it is an editorial change to the UFSAR or Technical Specifications.
B. Minor Calculation Change Forms do not change the status of the Parent Calculation Revision.

	Print/Sign		
Prepared by:	E. P. O'Brien	Date:	7/16/04
Reviewed by: *	E.G. Lind	Date:	7/16/04
Approved by	D.E. Yasi	Date:	7/20/04

* Where the original calculation was design verified, the reviewer signature confirms the latest design verification is still valid.

This IS a Quality Record -

Docket No. 50-271
DPS Exhibit #21
22 Pages

5. Description of Change

The NPSH evaluation of the RHR and CS pumps is performed, using the same methodology as CCN04 (Ref. 11) for short term DBA-Loss of Coolant Accident (LOCA) case and CCN05 (Ref. 12) for long term DBA-Loss of Coolant Accident (LOCA) case. This CCN uses the EPU suppression pool temperature/pressure data supplied in Reference 1. Note – all References contained in AP0017.07 (Attachment B)

Reason for Change:
This CCN provides additional results regarding the Residual Heat Removal (RHR) and Core Spray (CS) Pump NPSH resulting from the Extended Power Uprate (EPU), which was initially evaluated in CCN04 and subsequently in CCN05.

Specifically, this CCN06 updates the short-term and long-term post LOCA results based on revised containment temperature and pressure profiles provided in Reference 1. The revised containment profiles reflect that the Containment Spray Thermal Mixing Efficiency used to develop the Reference 1 input was increased to 100%. This increase in spray efficiency resulted in a slight reduction to the pressure profile. The containment spray thermal mixing efficiency utilized in CCN 5 was based on the containment air to steam mass ratio.

CCN06 remains based upon: (1) The long-term post LOCA time of 200,000 seconds, beyond which overpressure is no longer being required, and (2) the containment leakage of 1.5 weight percent per day, to reflect the recently submitted Alternate Source Term (AST) License Amendment Request (Ref: 5, ERC-2004-024)

Table of Contents

Note: Figure 4.1 is not contained in this CCN.

TOTAL PAGES: 22

1.0 Objective

The objective of this CCN is to update the CCN05 (Ref. 12) post LOCA Extended Power Uprate (EPU) evaluation of the adequacy of the available NPSH for the Residual Heat Removal (RHR) and Core Spray (CS) pumps. This includes identification of any change in the amount and duration of suppression pool (torus) overpressure required to maintain adequate NPSH.

Specifically, this CCN06 updates the post LOCA results based on revised containment temperature and pressure profiles provided in Reference 1. The revised containment profiles reflect that the Spray Efficiency used to develop the Reference 1 input is 100%, which resulted in a slight reduction to the pressure profile from CCN05.Additionally, Reference 1 provided revised short-term LOCA data. Therefore, Section 4.1 of CCN04 to VYC-0808 Rev 6 (Ref. 11) is updated to include this data, though the results will be shown to remain unchanged.

2.0 Methodology

General

The methodology for determining the NPSH available (NPSHa) for a given event and temperature is the same as that developed in VYC-0808 Rev 6 and presented in Table 1 of that calculation (Ref: 4). The NPSH required (NPSHr) is also per VYC-0808 Rev 6 and is discussed in detail in Section 4.0 of this CCN. The methodology for determining the pump suction strainer head loss during a LOCA, and the time dependent profile for required overpressure is the same as that developed in CCN04.

3.0 Assumptions

1. None made

4.0 Analysis

As stated in Section 2.0, the methodology for determining the NPSHa is the same as that developed in VYC-0808 Rev 6 and presented in Table 1 of that calculation (Ref: 4). The following terms are used in the evaluation.

NPSHa (ft) = net positive suction head available without overpressure credit
(14.7-Pg)(144vf)+Z-hf-hs-hd

where:

Z (ft) = suction elevation head
hf (ft) = suction line losses
hs (ft) = clean strainer losses
hd (ft) = strainer debris losses
Pg (psia) = vapor pressure @ torus temperature
vf (ft^3/lb) = specific volume @ torus temperature and pressure

Pg and vf are obtained from ASME Steam Tables 1967 Formulation (Ref: 7)

NPSHr (ft) = net positive suction head required.

It should be noted that the NPSH required data provided by the pump vendor, as documented in Figures 2.1-1 and 2.2-1 of Attachment 3 to calculation VYC-0808, is actually *Allowable Operating Periods @ NPSHa Specified.* Allowable hours of operation at specified NPSHa values are identified for a range of flows. For this CCN, the NPSHa specified in these Figures is taken as the NPSHr at a given operating time.

Q (gpm) = pump flow rate

OPR (psig) = Overpressure Required
(NPSHr – NPSHa)/(144*Vf)

For those profile points where there is inadequate NPSH, when considering the suppression pool pressure to be atmospheric (14.7 psia), OPR is the amount of suppression pool pressure required to make NPSHa (ft) equal to NPSHr (ft).

OPA (psig)– Overpressure Available
The suppression pool pressure available, above atmospheric, for a given event and time.

OPC (psig)– Overpressure Credit Taken
The overpressure credited in the evaluation of NPSH. Engineering judgement is used to select the credit to be greater than the OPR, by a reasonable amount, and less than the OPA.

Detailed discussion of the above terms is provided in the subsections that follow.

4.1 LOCA – Short Term

The temperature and pressure (T/P) profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-346R1 (Ref: 1). The short term data is provided from 0-600 seconds.

The evaluation of NPSH is documented in Table 4.1 using the peak pool temperature of 165.1°F which occurs at 600 seconds with a corresponding pool pressure of 17.64 psia. The peak temperature results in the largest vapor pressure and lowest NPSHa. Note that the temperature at lowest pool pressure is 161.2°F / 17.40psia. At this temperature the gain in vapor pressure more than offsets the reduction in pool pressure, therefore the 165.1°F case governs. The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for CS and RHR. Further discussion of selected terms is presented below.

Suction Elevation Head, Z
The values of Z for RHR and CS (12.30' and 12.47'respectively) as calculated in Section 3.5 of VYC-0808 are conservatively used in this evaluation. The suction elevation head is based on the water elevation in the torus. The EPU suppression pool water volume is slightly larger than the existing value used in VYC-0808, which would result in a slight increase in water elevation, and therefore Z.

A water volume comparison at maximum pool temperature is provided below:

	Pre-EPU	EPU
Ref:	(VYC-0808 Rev 6 Section 3.5)	(GE-VYNPS-AEP-346R1)
Short Term	76,800 cuft	79,390 cuft

Maximum Debris Losses (hd)

1 RHR: CCN #3 (Ref: 3) calculated the limiting head loss as 0.24 ft at 181.7°F and 7400 gpm. Note that this is a slight reduction from the head loss (0.33 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.33 ft at 173°F is used. (Case 1 of Tables 2 and 8 of Ref: 2).

2 RHR: The head loss is taken as .48 ft (Ref: 4) at 170°F (Case 2b of Tables 2 and 8 of Ref: 2) and 14200 gpm.

CS The head loss is conservatively taken as .32 ft (Ref: 4) at 173°F (Case 3d of Tables 2 and 8 of Ref: 2) and 4600 gpm.

Refer to Section 2.0 of CCN04 (Ref. 11) for application of head loss at temperatures other than those used in its calculation.

<u>NPRHr - CS</u>
Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 4600 gpm an allowable NPSH of 28.0 ft is acceptable between 0 and 7 hrs of operation.

<u>NPRHr - 1 RHR</u>
Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 7400 gpm an allowable NPSH of ~~23.8 ft~~ 23.9 is acceptable between 0 and 7 hrs of operation.

<u>NPRHr - 2 RHR</u>
With two RHR pumps operating at a total flow of 14,200 gpm this yields a flow of 7100 gpm per pump.
Also per Figure 2.1-1, the plot shows that at between 0 and 7 hrs of operation, an allowable NPSH of 23.5 ft is acceptable at 7000 gpm and 24.0 ft is acceptable at 7600 gpm.

Interpolating between plotted NPSH values of 23.5 ft @ 7000 gpm and 24.0 ft @ 7600 gpm yields 23.6 ft @ 7100 gpm.

The interpolation equation is developed as documented Section 2.2.2 of VYC-0808 Rev 6 and is 23.0+(Q-6400)/1200

4.1.1 Evaluation
As can been seen from Table 4.1, there is adequate NPSHa and overpressure is not required.

4.2 LOCA – Long Term

The temperature and pressure (T/P) profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-346R1 (Ref: 1). The long term data is provided from 0-864,000 seconds.

The evaluation of NPSH is documented in Table 4.2 using selected T/P points representing the long term profile of the suppression pool. The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for the T/P profile of CS and RHR. The evaluated long term flow rates of 7400 gpm (RHR) and 3500 gpm (CS) are consistent with calculation VYC-0808 Rev 6 (Ref: 4). Further discussion of selected terms is presented below.

<u>Suction Elevation Head, Z</u>
The values of Z for RHR and CS (12.40' and 12.57' respectively) as calculated in Section 3.5 of VYC-0808 are conservatively used in the evaluation. The suction elevation head is based on the water elevation in the torus. The EPU suppression pool water volume is slightly larger than the existing value used in VYC-0808, which would result in a slight increase in water elevation, and therefore Z.

A water volume comparison at maximum pool temperature is provided below:

	Pre-EPU	EPU
Ref:	(VYC-0808 Rev 6 Section 3.5)	(GE-VYNPS-AEP-346R1)
Long term	77,640 cuft	79,540 cuft

Maximum Debris Losses (hd)

1 RHR: CCN #3 (Ref: 3) calculated the limiting head loss as 0.24 ft at 181.7°F and 7400 gpm. Note that this is a slight reduction from the head loss (0.33 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.33 ft at 173°F is used. (Case 1 of Tables 2 and 8 of Ref: 2).

CS: Note that CCN #3 (Ref: 3) documents the up-to-date limiting head loss as 0.19 ft at 181.7°F and 3500 gpm. This is a slight reduction from the head loss (0.21 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.21 ft at 173°F is used. This is based on a conservative CS flow rate of 4000 gpm. (Case 3b of Tables 2 and 8 of Ref: 2).

NPRHr - CS

?

Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 3500 gpm the allowable NPSH increases between 7 and 20 hrs of operation and a value of 29.6 ft is acceptable beyond 20 hrs of operation. This maximum value is conservatively used for the entire long term period (>600 sec).

NPRHr - RHR

31.9

Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 7400 gpm the allowable NPSH increases between 7 and 100 hrs of operation and a value of 31.7 ft is acceptable beyond 100 hrs of operation. This maximum value is conservatively used for the entire long term period (>600 sec).

4.2.1 Evaluation

As can been seen from Figure 4.2 the overpressure required for RHR envelopes that required for CS and the overpressure varies continuously over time. In order to facilitate reporting and presentation of the overpressure required, an enveloping, stepped, overpressure credit is overlaid on Figure 4.2. Refer to Section 4.0 for discussion on selection of overpressure credit.

Though the long term flow rates are postulated at time 600 seconds (e.g. CS throttled down from 4600gpm to 3500gpm), it is not the intent of this calculation to imply at what time throttling should commence or how much throttling is required. This is a function of the time dependent NPSHr and pool temperature. This calculation conservatively evaluates the maximum NPSHr as occurring over the entire operating period (>600 sec). The actual NPSHr is lower between 0-7 hrs and increases after 7 hrs.

5.0 Summary of Results

NPSHa is rounded to the nearest 0.1ft and OPR, OPC, and OPA are rounded to the nearest 0.1psig.

5.1 LOCA - Short Term (0-600 sec):

NPSHa is adequate for both CS and RHR pumps without crediting overpressure. NPSHa shown below is at the peak temperature. The results remain unchanged from VYC-0808, Rev. 6, CCN04 (Ref. 11), Section 4.1, LOCA short-term results.

Pump	Total flow, gpm	NPSHr, ft	NPSHa, ft
CS	4,600	28.0	28.4
1RHR	7,400	23.8	31.1
2 RHR	14,200	23.6	28.8

5.2 LOCA - Long Term (>600 sec):

NPSHa is adequate for both CS and RHR pumps with an overpressure credit that varies over time, as shown in Fig. 4.2. NPSHa, OPR, OPC, OPA are shown below, at the peak temperature. The results remain unchanged from VYC-0808, Rev. 6, CCN05 (Ref. 12), LOCA long-term results, except for a slight reduction in Over Pressure Available (OPA).

Pump	Total flow, gpm	NPSHr, ft	NPSHa, ft	OPR, psig	OPC, psig	OPA, psig
CS	3,500	29.6	19.5	4.2	6.1	7.8
1RHR	7,400	31.7	19.6	5.1	6.1	7.8

6.0 Conclusions

Torus overpressure must be credited for 200,000 seconds for operating the RHR and CS pumps at EPU conditions for a long term DBA-Loss of Coolant Accident (LOCA) to achieve adequate NPSH available.

The results of this CCN will provide input to the PUSAR (Ref: 12) for the RHR and CS NPSH evaluation and will alter input to calculation VYC-1628 (Ref: 13) to address the need for crediting torus overpressure in the calculation of NPSH available. Note that calculation VYC-1628 may be superseded by GE EPU Analysis. The need for crediting torus overpressure in the RHR and CS NPSH evaluation, shall also be addressed in the SADBD (Ref: 14), UFSAR (Ref: 15), and system DBDs RHR (Ref: 16) and CS (Ref: 17).

Note that the changes to the UFSAR were originally proposed in CCN04 and CCN05 and are pending incorporation via the design change and licensing processes. This MCC simply updates the previous CCN04 and CCN05 performed under AP-0017. The UFSAR does not currently contain information on containment overpressure.

Note that use of overpressure credit must be approved by the NRC as part of EPU.

No specific 50.59 Screening/Evaluation is required for this CCN since all EPU design changes and associated 50.59 documentation will be part of VYDC-2003-008.

Table 4.1 LOCA – Short Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)

LOCA - Short Term

NPSHa = (14.7-Pg)(144Vf)+Z-hf-hs-hd
OPR = (NPSHr – NPSHa)/(144*Vf)
OPA = Over pressure available
OPC = Over pressure credited

Cross references:
Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
See Discussion in Section 4.0 of this CCN

<u>Short Term Flow Rate (gpm)</u>

1 RHR	Q = 7400	CS	Q = 4600	Table of 1 calc VYC-0808 Rev 6 (Ref: 4)
2 RHR	Q = 14200			

<u>Suction Line Losses (ft)</u>

1 RHR	hf = 4.77E-8*Q^2	CS	hf = 2.5E-7*Q^2	Section 3.7 of VYC-0808 Rev 6 (Ref: 4)
2 RHR	hf = 7.84E-8*(Q/2)^2			

<u>Clean Strainer Losses (ft)</u>

1 RHR	hs = 0.33	CS	hs = 0.51	Section 3.6 of VYC-0808 Rev 6 (Ref: 4)
2 RHR	hs = 1.22			

<u>Maximum Debris Losses (ft) @ >=base temperature</u> — See discussion in Section 4.1 of this CCN

1 RHR	hd = 0.33@173F	CS	hd = 0.32@173F	Ref: 2, 3, 4
2 RHR	hd = 0.48@170F			Ref: 2, 4

<u>Maximum Debris Losses (ft) @ < base temperature</u>

1 RHR	hd = .33*(173/T)	CS	hd = .32*(173/T)	See discussion in Section 2.0 of CCN04 (Ref. 11)
2 RHR	hd = .48*(170/T)			

where T = suppression pool temperature, F

<u>Elevation Head (ft)</u>

RHR	Z = 12.3	CS	Z = 12.47	See discussion in Section 4.1 of this CCN for conservatism.

<u>NPSHr (ft)</u>

1 RHR	NPSHr = 23.8	CS	NPSHr = 28.0	See discussion in Section 4.1 of this CCN
2 RHR	NPSHr = 23.6			

Short Term (After EPU) - Peak Torus Temperature - 1.5 wt. % Containment Leakage & 100% Spray Efficiency

Pump(s)	Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)	OPA (psig)	OPC (psig)
CS	600	165.1	17.64	5.349	0.016423	12.47	5.29	0.51	0.34	28.44	28.00	0.00	2.94	0.00
1 RHR	600	165.1	17.64	5.349	0.016423	12.30	2.61	0.33	0.35	31.12	23.80	0.00	2.94	0.00
2 RHR	600	165.1	17.64	5.349	0.016423	12.30	3.95	1.22	0.49	28.75	23.60	0.00	2.94	0.00

Table 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)

LOCA - Long Term

NPSHa = (14.7-Pg)(144Vf)+Z-hf-hs-hd
OPR = Over pressure required (NPSHr – NPSHa)/(144*Vf)
OPA = Over pressure available
OPC = Over pressure credited

Cross references:
Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
See Discussion in Section 4.0 of this CCN

Long Term Flow Rate (gpm)				
1 RHR	Q = 7400	CS	Q = 3500	Table of 1 calc VYC-0808 Rev 6 (Ref: 4)
Suction Line Losses (ft)				
1 RHR	hf = 4.77E-8*Q^2	CS	hf = 2.5E-7*Q^2	Section 3.7 of VYC-0808 Rev 6 (Ref: 4)
Clean Strainer Losses (ft)				
1 RHR	hs = 0.33	CS	hs = .38*(Q/4000)^2 for Q<= 4000	Section 3.6 of VYC-0808 Rev 6 (Ref: 4)
Maximum Debris Losses (ft) @ >= 173F				
1 RHR	hd = 0.33	CS	hd = 0.21	See discussion in Section 4.2 of this CCN
Maximum Debris Losses (ft) @ < 173F				
1 RHR	hd = .33*(173/T)	CS	hd = .21*(173/T)	See discussion in Section 2.0 of CCN04 (Ref. 11)
where T = suppression pool temperature, F				
Elevation Head (ft)				
RHR	Z = 12.4	CS	Z = 12.57	See discussion in Section 4.2 of this CCN for conservatism.
NPSHr (ft)				
1 RHR	NPSHr = 31.7	CS	NPSHr = 29.6	See discussion in Section 4.2 of this CCN

31.9?

Table 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)

CS - Long Term (After EPU) 1.5 wt. % Containment Leakage & 100% Spray Efficiency

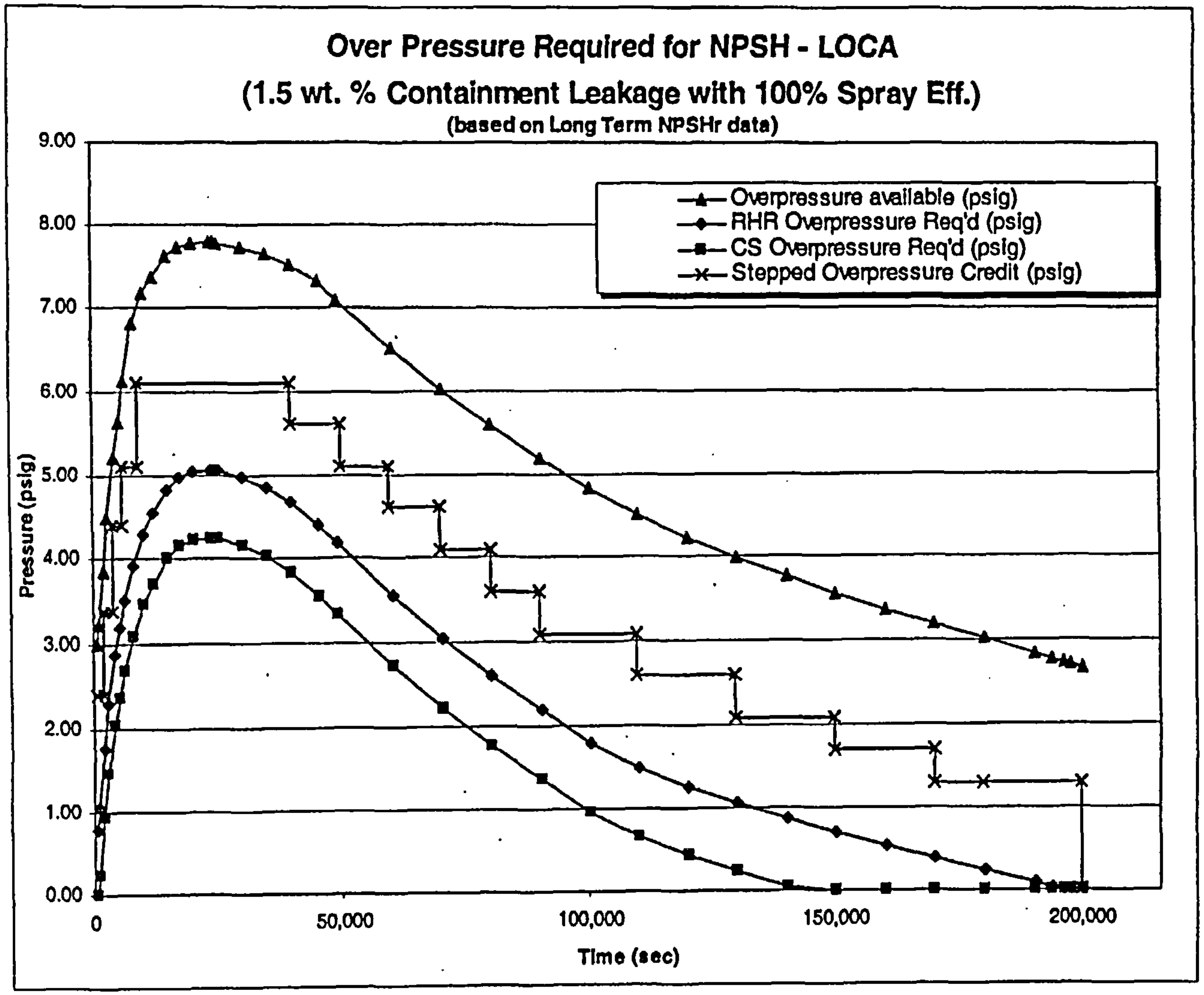
Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	CS NPSHa (ft)	CS NPSHr (ft)	CS OPR (psig)	OPA (psig)	OPC (psig)
786	169.7	17.71	5.951	0.016449	12.57	3.06	0.29	0.21	29.73	29.60	0.00	3.01	2.40
1,098	171.8	17.94	6.245	0.016461	12.57	3.06	0.29	0.21	29.05	29.60	0.23	3.24	2.40
2,033	176.6	18.57	6.962	0.016489	12.57	3.06	0.29	0.21	27.38	29.60	0.94	3.87	3.40
2,962	180.0	19.17	7.511	0.016509	12.57	3.06	0.29	0.21	26.10	29.60	1.47	4.47	3.40
4,196	183.4	19.90	8.096	0.016530	12.57	3.06	0.29	0.21	24.73	29.60	2.05	5.20	4.40
5,125	185.2	20.34	8.420	0.016541	12.57	3.06	0.29	0.21	23.96	29.60	2.37	5.64	4.40
6,275	187.0	20.82	8.756	0.016552	12.57	3.06	0.29	0.21	23.17	29.60	2.70	6.12	5.10
8,036	189.1	21.50	9.161	0.016566	12.57	3.06	0.29	0.21	22.22	29.60	3.09	6.80	5.10
10,220	191.0	21.86	9.541	0.016578	12.57	3.06	0.29	0.21	21.32	29.60	3.47	7.16	6.10
12,094	192.2	22.06	9.788	0.016585	12.57	3.06	0.29	0.21	20.74	29.60	3.71	7.36	6.10
15,170	193.6	22.31	10.083	0.016594	12.57	3.06	0.29	0.21	20.04	29.60	4.00	7.61	6.10
17,669	194.3	22.43	10.233	0.016599	12.57	3.06	0.29	0.21	19.68	29.60	4.15	7.73	6.10
20,156	194.6	22.46	10.298	0.016601	12.57	3.06	0.29	0.21	19.53	29.60	4.21	7.76	6.10
23,812	194.7	22.48	10.320	0.016601	12.57	3.06	0.29	0.21	19.48	29.60	4.23	7.78	6.10
24,495	194.7	22.48	10.320	0.016601	12.57	3.06	0.29	0.21	19.48	29.60	4.23	7.78	6.10
25,120	194.7	22.47	10.320	0.016601	12.57	3.06	0.29	0.21	19.48	29.60	4.23	7.77	6.10
30,095	194.3	22.42	10.233	0.016599	12.57	3.06	0.29	0.21	19.68	29.60	4.15	7.72	6.10
35,065	193.7	22.33	10.104	0.016595	12.57	3.06	0.29	0.21	19.99	29.60	4.02	7.63	6.10
40,020	192.8	22.20	9.914	0.016589	12.57	3.06	0.29	0.21	20.44	29.60	3.83	7.50	5.60
45,637	191.5	22.01	9.644	0.016581	12.57	3.06	0.29	0.21	21.08	29.60	3.57	7.31	5.60
49,406	190.4	21.78	9.420	0.016574	12.57	3.06	0.29	0.21	21.61	29.60	3.35	7.08	5.60
60,551	187.2	21.21	8.794	0.016554	12.57	3.06	0.29	0.21	23.09	29.60	2.73	6.51	4.60
70,342	184.4	20.72	8.275	0.016536	12.57	3.06	0.29	0.21	24.31	29.60	2.22	6.02	4.10
80,342	181.8	20.28	7.816	0.016520	12.57	3.06	0.29	0.21	25.38	29.60	1.77	5.58	3.60
90,340	179.3	19.89	7.395	0.016505	12.57	3.06	0.29	0.21	26.37	29.60	1.36	5.19	3.10
100,340	176.8	19.52	6.994	0.016490	12.57	3.06	0.29	0.21	27.30	29.60	0.97	4.82	3.10
110,340	174.8	19.20	6.686	0.016478	12.57	3.06	0.29	0.21	28.02	29.60	0.66	4.50	2.60
120,306	173.2	18.93	6.447	0.016469	12.57	3.06	0.29	0.21	28.58	29.60	0.43	4.23	2.60
130,302	171.8	18.69	6.245	0.016461	12.57	3.06	0.29	0.21	29.05	29.60	0.23	3.99	2.10
140,302	170.4	18.47	6.048	0.016453	12.57	3.06	0.29	0.21	29.51	29.60	0.04	3.77	2.10
150,302	169.1	18.27	5.870	0.016445	12.57	3.06	0.29	0.21	29.92	29.60	0.00	3.57	1.70
160,302	167.8	18.07	5.696	0.016438	12.57	3.06	0.29	0.22	30.31	29.60	0.00	3.37	1.70
170,302	166.6	17.90	5.539	0.016431	12.57	3.06	0.29	0.22	30.67	29.60	0.00	3.20	1.30
180,302	165.3	17.72	5.374	0.016424	12.57	3.06	0.29	0.22	31.05	29.60	0.00	3.02	1.30
190,302	164.1	17.54	5.225	0.016417	12.57	3.06	0.29	0.22	31.40	29.60	0.00	2.84	1.30
194,052	163.6	17.47	5.164	0.016414	12.57	3.06	0.29	0.22	31.54	29.60	0.00	2.77	1.30
196,552	163.3	17.43	5.127	0.016413	12.57	3.06	0.29	0.22	31.62	29.60	0.00	2.73	1.30
197,802	163.2	17.41	5.115	0.016412	12.57	3.06	0.29	0.22	31.65	29.60	0.00	2.71	1.30
200,302	162.9	17.37	5.079	0.016411	12.57	3.06	0.29	0.22	31.73	29.60	0.00	2.67	0.00

Table 4.2 - LOCA – Long Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)

RHR - Long Term (After EPU) 1.5 wt. % Containment Leakage & 100% Spray Efficiency

Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	RHR NPSHa (ft)	RHR NPSHr (ft)	RHR OPR (psig)	OPA (psig)	OPC (psig)
786	169.7	17.71	5.951	0.016449	12.40	2.61	0.33	0.34	29.84	31.70	0.78	3.01	2.40
1,098	171.8	17.94	6.245	0.016461	12.40	2.61	0.33	0.33	29.17	31.70	1.07	3.24	2.40
2,033	176.6	18.57	6.962	0.016489	12.40	2.61	0.33	0.33	27.50	31.70	1.77	3.87	3.40
2,962	180.0	19.17	7.511	0.016509	12.40	2.61	0.33	0.33	26.22	31.70	2.31	4.47	3.40
4,196	183.4	19.90	8.096	0.016530	12.40	2.61	0.33	0.33	24.85	31.70	2.88	5.20	4.40
5,125	185.2	20.34	8.420	0.016541	12.40	2.61	0.33	0.33	24.09	31.70	3.20	5.64	4.40
6,275	187.0	20.82	8.756	0.016552	12.40	2.61	0.33	0.33	23.30	31.70	3.53	6.12	5.10
8,036	189.1	21.50	9.161	0.016566	12.40	2.61	0.33	0.33	22.34	31.70	3.92	6.80	5.10
10,220	191.0	21.86	9.541	0.016578	12.40	2.61	0.33	0.33	21.44	31.70	4.30	7.16	6.10
12,094	192.2	22.06	9.788	0.016585	12.40	2.61	0.33	0.33	20.86	31.70	4.54	7.36	6.10
15,170	193.6	22.31	10.083	0.016594	12.40	2.61	0.33	0.33	20.16	31.70	4.83	7.61	6.10
17,669	194.3	22.43	10.233	0.016599	12.40	2.61	0.33	0.33	19.81	31.70	4.98	7.73	6.10
20,156	194.6	22.46	10.298	0.016601	12.40	2.61	0.33	0.33	19.65	31.70	5.04	7.76	6.10
23,812	194.7	22.48	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	7.78	6.10
24,495	194.7	22.48	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	7.78	6.10
25,120	194.7	22.47	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	7.77	6.10
30,095	194.3	22.42	10.233	0.016599	12.40	2.61	0.33	0.33	19.81	31.70	4.98	7.72	6.10
35,065	193.7	22.33	10.104	0.016595	12.40	2.61	0.33	0.33	20.11	31.70	4.85	7.63	6.10
40,020	192.8	22.20	9.914	0.016589	12.40	2.61	0.33	0.33	20.56	31.70	4.66	7.50	5.60
45,637	191.5	22.01	9.644	0.016581	12.40	2.61	0.33	0.33	21.20	31.70	4.40	7.31	5.60
49,406	190.4	21.78	9.420	0.016574	12.40	2.61	0.33	0.33	21.73	31.70	4.18	7.08	5.60
60,551	187.2	21.21	8.794	0.016554	12.40	2.61	0.33	0.33	23.21	31.70	3.56	6.51	4.60
70,342	184.4	20.72	8.275	0.016536	12.40	2.61	0.33	0.33	24.43	31.70	3.05	6.02	4.10
80,342	181.8	20.28	7.816	0.016520	12.40	2.61	0.33	0.33	25.50	31.70	2.60	5.58	3.60
90,340	179.3	19.89	7.395	0.016505	12.40	2.61	0.33	0.33	26.49	31.70	2.19	5.19	3.10
100,340	176.8	19.52	6.994	0.016490	12.40	2.61	0.33	0.33	27.43	31.70	1.80	4.82	3.10
110,340	174.8	19.20	6.686	0.016478	12.40	2.61	0.33	0.33	28.14	31.70	1.50	4.50	2.60
120,306	173.2	18.93	6.447	0.016469	12.40	2.61	0.33	0.33	28.70	31.70	1.26	4.23	2.60
130,302	171.8	18.69	6.245	0.016461	12.40	2.61	0.33	0.33	29.17	31.70	1.07	3.99	2.10
140,302	170.4	18.47	6.048	0.016453	12.40	2.61	0.33	0.34	29.62	31.70	0.88	3.77	2.10
150,302	169.1	18.27	5.870	0.016445	12.40	2.61	0.33	0.34	30.03	31.70	0.71	3.57	1.70
160,302	167.8	18.07	5.696	0.016438	12.40	2.61	0.33	0.34	30.43	31.70	0.54	3.37	1.70
170,302	166.6	17.90	5.539	0.016431	12.40	2.61	0.33	0.34	30.79	31.70	0.38	3.20	1.30
180,302	165.3	17.72	5.374	0.016424	12.40	2.61	0.33	0.35	31.16	31.70	0.23	3.02	1.30
190,302	164.1	17.54	5.225	0.016417	12.40	2.61	0.33	0.35	31.51	31.70	0.08	2.84	1.30
194,052	163.6	17.47	5.164	0.016414	12.40	2.61	0.33	0.35	31.65	31.70	0.02	2.77	1.30
196,552	163.3	17.43	5.127	0.016413	12.40	2.61	0.33	0.35	31.73	31.70	0.00	2.73	1.30
197,802	163.2	17.41	5.115	0.016412	12.40	2.61	0.33	0.35	31.76	31.70	0.00	2.71	1.30
200,302	162.9	17.37	5.079	0.016411	12.40	2.61	0.33	0.35	31.84	31.70	0.00	2.67	0.00

Figure 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)



OPC overpress credit	
(sec)	(psig)
601	2.4
2000	2.4
2001	3.4
4000	3.4
4001	4.4
6000	4.4
6001	5.1
9000	5.1
9001	6.1
40000	6.1
40001	5.6
50000	5.6
50001	5.1
60000	5.1
60001	4.6
70000	4.6
70001	4.1
80000	4.1
80001	3.6
90000	3.6
90001	3.1
110000	3.1
110001	2.6
130000	2.6
130001	2.1
150000	2.1
150001	1.7
170000	1.7
170001	1.3
180000	1.3
200000	1.3
200001	0

Excel Verification Sample Calculation

Table 4.1 CS Verification (Line 40):

Cell G40 = $F$30
= 12.47

Cell H40 = 0.00000025*$F$9^2
= 0.00000025*4600^2
= 5.29

Cell I40 = $F$17
= 0.51

Cell J40 = ROUND(IF(C40<173,0.32*(173/C40),0.32),2)
= 165.1<173 = True
= 0.32*(173/C40)
= 0.32*(173/165.1)
= 0.34

Cell K40 =+((14.7-E40)*144*F40)+G40-H40-I40-J40
= ((14.7-5.349)*144*0.016423)+12.47-5.29-0.51-0.34
= 28.44

Cell L40 = $F$33
= 28.00

Cell M40 =IF((+L40-K40)/(144*F40)>0,(+L40-K40)/(144*F40),0)
=(28.00-28.44)/(144*0.016423) = -0.186<0, False
= 0

Cell N40 = +D40-14.7
= 17.64-14.7
= 2.94

Table 4.2 CS Table Verification (Line 50):

Cell F50 = $F$25
= 12.57

Cell G50 = 0.00000025*$F$9^2
= 0.00000025*3500^2
= 3.06

Cell H50 = 0.38*(($F$9/4000)^2)
= 0.38*((3500/4000)^2)
= 0.29

Cell I50 = ROUND(IF(B50<173,$F$18*(173/B50),$F$18),2)
194.3<173 = False (For True outcome see below, Cell I72)
= $F$18
= 0.21

Cell I72 = ROUND(IF(B72<173,$F$18*(173/B72),$F$18),2)
= 162.9<173 = True
= $F$18*(173/B72)
= 0.21*(173/162.9)
= 0.22

Cell J50 = +((14.7-D50)*144*E50)+F50-G50-H50-I50
= ((14.7-10.233)*144*0.016599)+12.57-3.06-0.29-0.21
= 19.687 [Worksheet shows 19.68 - Check OK - difference attributed to significant figures used in hand calc vs Excel]

Cell K50 = $F$28
= 29.6

Cell L50 = IF((+K50-J50)/(144*E50)>0,(+K50-J50)/(144*E50),0)
(29.6-19.68)/(144*0.016599) = 4.15>0 True (For False outcome see below, Cell L34)
= 4.15

Cell L34 = IF((+K34-J34)/(144*E34)>0,(+K34-J34)/(144*E34),0)
= (29.6-29.73)/(144*0.016449) = -0.055>0 False
= 0

Cell M50 = +C50-14.7
= 22.42-14.7
= 7.72

Table 4.2 RHR Table Verification (Line 78):

Cell F78 = $C$25
= 12.40

Cell G78 = 0.0000000477*$C$9^2
= 0.0000000477*7400^2
= 2.61

Cell H78 = $C$15
= 0.33

Cell I78 = ROUND(IF(B78<173,$C$18*(173/B78),$C$18),2)
169.7<173 = True (For False outcome see below, Cell I81)
= $C$18*(173/B78)
= 0.33*(173/169.7)
= 0.34

Cell I81 = ROUND(IF(B81<173,$C$18*(173/B81),$C$18),2)
180<173 = False
= $C$18
= 0.33

Cell J78 = +((14.7-D78)*144*E78)+F78-G78-H78-I78
= ((14.7-5.951)*144*0.016449)+12.40-2.61-0.33-0.34
= 29.84

Cell K78 = $C$28
= 31.7

Cell L78 = IF((+K78-J78)/(144*E78)>0,(+K78-J78)/(144*E78),0)
(31.7-29.84)/(144*0.016449)=0.785>0 True (For False outcome see below, Cell L116)
= 0.785 [Worksheet shows 0.78 - Check OK - difference attributed to significant figures used in hand calc vs Excel]

Cell L116 = IF((+K116-J116)/(144*E116)>0,(+K116-J116)/(144*E116),0)
= (31.7-31.84)/(144*0.016411) = -0.059>0 False
= 0

Cell M78 = +C78-14.7
= 17.71-14.7
= 3.01

	A	B	C	D	E	F	G	H
1	LOCA - Short Term							Cross reference:
2	NPSHa =	(14.7-Pg)(144Vf)+Z-hf-hs-hd						Section 2.3 of VYC-0808 Rev 6 (Ref. 4)
3	OPR =	(NPSHr - NPSHa)/(144*Vf)						See Discussion in Section 4.0 of this CCN
4	OPA =	Over pressure available						
5	OPC =	Over pressure credited						
6								
7								
8	Short Term Flow Rate (gpm)							
9	1 RHR		Q = 7400		CS	Q = 4600		Table of 1 calc VYC-0808 Rev 6 (Ref. 4)
10	2 RHR		Q = 14200					
11								
12	Suction Line Losses (ft)							
13	1 RHR		hf = 4.77E-8*Q^2		CS	hf = 2.5E-7*Q^2		Section 3.7 of VYC-0808 Rev 6 (Ref. 4)
14	2 RHR		hf = 7.84E-8*(Q/2)^2					
15								
16	Clean Strainer Losses (ft)							
17	1 RHR		hs = 0.33		CS	hs = 0.51		Section 3.6 of VYC-0808 Rev 6 (Ref. 4)
18	2 RHR		hs = 1.22					
19								
20	Maximum Debris Losses (ft) @ ≥ base temperature							See discussion in Section 4.1 of this CCN
21	1 RHR		hd = 0.33 @ 173F		CS	hd = 0.32 @ 173F		Ref. 2, 3, 4
22	2 RHR		hd = 0.46 @ 170F					Ref. 2, 4
23								
24	Maximum Debris Losses (ft) @ < base temperature							
25	1 RHR		hd = .33*(173/T)		CS	hd = .32*(173/T)		See discussion in Section 2.8 of CCN04 (Ref. 11)
26	2 RHR		hd = .46*(170/T)					
27	where T = suppression pool temperature, F							
28								
29	Elevation Head (ft)							
30	RHR		Z = 12.3		CS	Z = 12.47		See discussion in Section 4.1 of this CCN for conservatism.
31								
32	NPSHr (ft)							
33	1 RHR		NPSHr = 25.8		CS	NPSHr = 35		See discussion in Section 4.1 of this CCN
34	2 RHR		NPSHr = 23.6					
35								

36 Short Term (After EPU) - Peak Torus Temperature - 1.6 wt. % Containment Leakage & 100% Spray Efficiency

37–39	Pump(s)	Time (sec)	OE Pool Temp (F)	OE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)	OPA (psig)	OPC (psig)
40	CS	600	165.1	17.64	5.348	0.016423	=$F$30	=0.00000025*$F$9^2	=$F$17	=ROUND(IF(C40>173,0.32*(173/C40),0.32),2)	=((14.7-E40)*144*F40)+G40-H40-I40-J40	=$F$33	=IF(((L40-K40)/(144*F40))>0,((L40-K40)/(144*F40)),0)	=D40-14.7	0
41	1 RHR	600	165.1	17.64	5.348	0.016423	=$C$30	=0.000000477*$C$9^2	=$C$17	=ROUND(IF(C41>173,0.33*(173/C41),0.33),2)	=((14.7-E41)*144*F41)+G41-H41-I41-J41	=$C$33	=IF(((L41-K41)/(144*F41))>0,((L41-K41)/(144*F41)),0)	=D41-14.7	0
42	2 RHR	600	165.1	17.64	5.348	0.016423	=$C$30	=0.0000000784*($C$10/2)^2	=$C$18	=ROUND(IF(C42>170,0.46*(170/C42),0.46),2)	=((14.7-E42)*144*F42)+G42-H42-I42-J42	=$C$34	=IF(((L42-K42)/(144*F42))>0,((L42-K42)/(144*F42)),0)	=D42-14.7	0

	A	B	C	D	E	F	G	H	I	J	K	L	M	N
1	LOCA - Long Term							Cross references:						
2	NPSHa =	(14.7-Pg)(144Vf)+Z-hf-hs-hd						Section 2.3 of VYC-0808 Rev 6 (Ref. 4)						
3	OPR =	Over pressure required (NPSHr - NPSHa)/(144*Vf)						See Discussion in Section 4.8 of this CCN						
4	OPA =	Over pressure available												
5	OPC =	Over pressure credited												
6														
7														
8	Long Term Flow Rate (gpm)													
9	1 RHR	Q = 7400			CS	Q = 3600		Table of 1 calc VYC-0808 Rev 6 (Ref. 4)						
10														
11	Suction Line Losses (ft)													
12	1 RHR	hf = 4.77E-8*Q^2			CS	hf = 2.5E-7*Q^2		Section 3.7 of VYC-0808 Rev 6 (Ref. 4)						
13														
14	Clean Strainer Losses (ft)													
15	1 RHR	hs = 0.33			CS	hs = .38*(Q/4000)^2 for Q<= 4000		Section 3.8 of VYC-0808 Rev 6 (Ref. 4)						
16														
17	Maximum Debris Losses (ft) @ >= 173F													
18	1 RHR	hd = 0.33			CS	hd = 0.21		See discussion in Section 4.2 of this CCN						
19														
20	Maximum Debris Losses (ft) @ < 173F													
21	1 RHR	hd = .33*(173/T)			CS	hd = .21*(173/T)		See discussion in Section 2.8 of CCN04 (Ref. 11)						
22	where T = suppression pool temperature, F													
23														
24	Elevation Head (ft)													
25	RHR	Z = 12.4			CS	Z = 12.57		See discussion in Section 4.2 of this CCN for conservatism.						
26														
27	NPSHr (ft)													
28	1 RHR	NPSHr = 31.7			CS	NPSHr = 28.6		See discussion in Section 4.2 of this CCN						
29														
30	CS - Long Term (After EPU) 1.5 wt. % Containment Leakage & 100% Spray Efficiency													
31-33	Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	CS NPSHa (ft)	CS NPSHr (ft)	CS OPR (psig)	OPA (psig)	OPC (psig)
34	785.6045	168.7	17.71	6.951	0.016448	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B34<173,$F$18*(173/B34),$F$18),2)	=((14.7-D34)*144*E34)+F34-G34-H34-I34	=$F$28	=IF((K34-J34)/(144*E34)>0,(K34-J34)/(144*E34),0)	=C34-14.7	=VLOOKUP(A34,$O$122:$K$153,2)
35	1087.869	171.8	17.84	6.246	0.016461	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B35<173,$F$18*(173/B35),$F$18),2)	=((14.7-D35)*144*E35)+F35-G35-H35-I35	=$F$28	=IF((K35-J35)/(144*E35)>0,(K35-J35)/(144*E35),0)	=C35-14.7	=VLOOKUP(A35,$O$122:$K$153,2)
36	2033.366	176.8	18.57	6.962	0.016480	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B36<173,$F$18*(173/B36),$F$18),2)	=((14.7-D36)*144*E36)+F36-G36-H36-I36	=$F$28	=IF((K36-J36)/(144*E36)>0,(K36-J36)/(144*E36),0)	=C36-14.7	=VLOOKUP(A36,$O$122:$K$153,2)
37	2982.323	180	19.17	7.611	0.016500	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B37<173,$F$18*(173/B37),$F$18),2)	=((14.7-D37)*144*E37)+F37-G37-H37-I37	=$F$28	=IF((K37-J37)/(144*E37)>0,(K37-J37)/(144*E37),0)	=C37-14.7	=VLOOKUP(A37,$O$122:$K$153,2)
38	4196.48	183.4	19.9	8.086	0.01653	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B38<173,$F$18*(173/B38),$F$18),2)	=((14.7-D38)*144*E38)+F38-G38-H38-I38	=$F$28	=IF((K38-J38)/(144*E38)>0,(K38-J38)/(144*E38),0)	=C38-14.7	=VLOOKUP(A38,$O$122:$K$153,2)
39	5175.042	186.2	20.34	8.42	0.016541	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B39<173,$F$18*(173/B39),$F$18),2)	=((14.7-D39)*144*E39)+F39-G39-H39-I39	=$F$28	=IF((K39-J39)/(144*E39)>0,(K39-J39)/(144*E39),0)	=C39-14.7	=VLOOKUP(A39,$O$122:$K$153,2)
40	6275.292	187	20.82	8.756	0.016552	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B40<173,$F$18*(173/B40),$F$18),2)	=((14.7-D40)*144*E40)+F40-G40-H40-I40	=$F$28	=IF((K40-J40)/(144*E40)>0,(K40-J40)/(144*E40),0)	=C40-14.7	=VLOOKUP(A40,$O$122:$K$153,2)
41	8035.636	189.1	21.8	9.161	0.016566	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B41<173,$F$18*(173/B41),$F$18),2)	=((14.7-D41)*144*E41)+F41-G41-H41-I41	=$F$28	=IF((K41-J41)/(144*E41)>0,(K41-J41)/(144*E41),0)	=C41-14.7	=VLOOKUP(A41,$O$122:$K$153,2)
42	10219.09	191	21.66	9.541	0.016579	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B42<173,$F$18*(173/B42),$F$18),2)	=((14.7-D42)*144*E42)+F42-G42-H42-I42	=$F$28	=IF((K42-J42)/(144*E42)>0,(K42-J42)/(144*E42),0)	=C42-14.7	=VLOOKUP(A42,$O$122:$K$153,2)
43	12093.67	192.2	22.06	9.788	0.016586	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B43<173,$F$18*(173/B43),$F$18),2)	=((14.7-D43)*144*E43)+F43-G43-H43-I43	=$F$28	=IF((K43-J43)/(144*E43)>0,(K43-J43)/(144*E43),0)	=C43-14.7	=VLOOKUP(A43,$O$122:$K$153,2)
44	15149.8	193.6	22.31	10.063	0.016594	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B44<173,$F$18*(173/B44),$F$18),2)	=((14.7-D44)*144*E44)+F44-G44-H44-I44	=$F$28	=IF((K44-J44)/(144*E44)>0,(K44-J44)/(144*E44),0)	=C44-14.7	=VLOOKUP(A44,$O$122:$K$153,2)
45	17689.2	194.3	22.43	10.233	0.016599	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B45<173,$F$18*(173/B45),$F$18),2)	=((14.7-D45)*144*E45)+F45-G45-H45-I45	=$F$28	=IF((K45-J45)/(144*E45)>0,(K45-J45)/(144*E45),0)	=C45-14.7	=VLOOKUP(A45,$O$122:$K$153,2)
46	20156.38	194.6	22.46	10.298	0.016601	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B46<173,$F$18*(173/B46),$F$18),2)	=((14.7-D46)*144*E46)+F46-G46-H46-I46	=$F$28	=IF((K46-J46)/(144*E46)>0,(K46-J46)/(144*E46),0)	=C46-14.7	=VLOOKUP(A46,$O$122:$K$153,2)
47	23411.7	194.7	22.49	10.32	0.016601	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B47<173,$F$18*(173/B47),$F$18),2)	=((14.7-D47)*144*E47)+F47-G47-H47-I47	=$F$28	=IF((K47-J47)/(144*E47)>0,(K47-J47)/(144*E47),0)	=C47-14.7	=VLOOKUP(A47,$O$122:$K$153,2)
48	24494.98	194.7	22.49	10.32	0.016601	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B48<173,$F$18*(173/B48),$F$18),2)	=((14.7-D48)*144*E48)+F48-G48-H48-I48	=$F$28	=IF((K48-J48)/(144*E48)>0,(K48-J48)/(144*E48),0)	=C48-14.7	=VLOOKUP(A48,$O$122:$K$153,2)
49	25119.98	194.7	22.47	10.32	0.016601	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B49<173,$F$18*(173/B49),$F$18),2)	=((14.7-D49)*144*E49)+F49-G49-H49-I49	=$F$28	=IF((K49-J49)/(144*E49)>0,(K49-J49)/(144*E49),0)	=C49-14.7	=VLOOKUP(A49,$O$122:$K$153,2)
50	30094.54	194.3	22.42	10.273	0.016600	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B50<173,$F$18*(173/B50),$F$18),2)	=((14.7-D50)*144*E50)+F50-G50-H50-I50	=$F$28	=IF((K50-J50)/(144*E50)>0,(K50-J50)/(144*E50),0)	=C50-14.7	=VLOOKUP(A50,$O$122:$K$153,2)
51	36064.61	193.7	22.33	10.104	0.016595	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B51<173,$F$18*(173/B51),$F$18),2)	=((14.7-D51)*144*E51)+F51-G51-H51-I51	=$F$28	=IF((K51-J51)/(144*E51)>0,(K51-J51)/(144*E51),0)	=C51-14.7	=VLOOKUP(A51,$O$122:$K$153,2)
52	40000.36	192.8	22.2	9.914	0.016588	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B52<173,$F$18*(173/B52),$F$18),2)	=((14.7-D52)*144*E52)+F52-G52-H52-I52	=$F$28	=IF((K52-J52)/(144*E52)>0,(K52-J52)/(144*E52),0)	=C52-14.7	=VLOOKUP(A52,$O$122:$K$153,2)
53	45437.23	191.5	22.01	9.644	0.016581	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B53<173,$F$18*(173/B53),$F$18),2)	=((14.7-D53)*144*E53)+F53-G53-H53-I53	=$F$28	=IF((K53-J53)/(144*E53)>0,(K53-J53)/(144*E53),0)	=C53-14.7	=VLOOKUP(A53,$O$122:$K$153,2)
54	49408.11	190.4	21.78	9.42	0.016574	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B54<173,$F$18*(173/B54),$F$18),2)	=((14.7-D54)*144*E54)+F54-G54-H54-I54	=$F$28	=IF((K54-J54)/(144*E54)>0,(K54-J54)/(144*E54),0)	=C54-14.7	=VLOOKUP(A54,$O$122:$K$153,2)
55	60650.64	187.2	21.21	8.794	0.016554	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B55<173,$F$18*(173/B55),$F$18),2)	=((14.7-D55)*144*E55)+F55-G55-H55-I55	=$F$28	=IF((K55-J55)/(144*E55)>0,(K55-J55)/(144*E55),0)	=C55-14.7	=VLOOKUP(A55,$O$122:$K$153,2)
56	70342.79	184.4	20.72	8.275	0.016536	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B56<173,$F$18*(173/B56),$F$18),2)	=((14.7-D56)*144*E56)+F56-G56-H56-I56	=$F$28	=IF((K56-J56)/(144*E56)>0,(K56-J56)/(144*E56),0)	=C56-14.7	=VLOOKUP(A56,$O$122:$K$153,2)
57	80342.79	181.8	20.26	7.818	0.01652	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B57<173,$F$18*(173/B57),$F$18),2)	=((14.7-D57)*144*E57)+F57-G57-H57-I57	=$F$28	=IF((K57-J57)/(144*E57)>0,(K57-J57)/(144*E57),0)	=C57-14.7	=VLOOKUP(A57,$O$122:$K$153,2)
58	90336.91	179.3	19.89	7.386	0.016505	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B58<173,$F$18*(173/B58),$F$18),2)	=((14.7-D58)*144*E58)+F58-G58-H58-I58	=$F$28	=IF((K58-J58)/(144*E58)>0,(K58-J58)/(144*E58),0)	=C58-14.7	=VLOOKUP(A58,$O$122:$K$153,2)
59	100330.8	176.8	18.52	6.994	0.01649	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B59<173,$F$18*(173/B59),$F$18),2)	=((14.7-D59)*144*E59)+F59-G59-H59-I59	=$F$28	=IF((K59-J59)/(144*E59)>0,(K59-J59)/(144*E59),0)	=C59-14.7	=VLOOKUP(A59,$O$122:$K$153,2)
60	110330.7	174.8	19.2	6.606	0.016478	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B60<173,$F$18*(173/B60),$F$18),2)	=((14.7-D60)*144*E60)+F60-G60-H60-I60	=$F$28	=IF((K60-J60)/(144*E60)>0,(K60-J60)/(144*E60),0)	=C60-14.7	=VLOOKUP(A60,$O$122:$K$153,2)
61	120306.5	173.2	18.83	6.447	0.016468	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B61<173,$F$18*(173/B61),$F$18),2)	=((14.7-D61)*144*E61)+F61-G61-H61-I61	=$F$28	=IF((K61-J61)/(144*E61)>0,(K61-J61)/(144*E61),0)	=C61-14.7	=VLOOKUP(A61,$O$122:$K$153,2)
62	130302	171.8	18.69	6.245	0.016461	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B62<173,$F$18*(173/B62),$F$18),2)	=((14.7-D62)*144*E62)+F62-G62-H62-I62	=$F$28	=IF((K62-J62)/(144*E62)>0,(K62-J62)/(144*E62),0)	=C62-14.7	=VLOOKUP(A62,$O$122:$K$153,2)
63	140302	170.4	18.47	6.046	0.016453	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B63<173,$F$18*(173/B63),$F$18),2)	=((14.7-D63)*144*E63)+F63-G63-H63-I63	=$F$28	=IF((K63-J63)/(144*E63)>0,(K63-J63)/(144*E63),0)	=C63-14.7	=VLOOKUP(A63,$O$122:$K$153,2)
64	150302	169.1	18.27	5.87	0.016445	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B64<173,$F$18*(173/B64),$F$18),2)	=((14.7-D64)*144*E64)+F64-G64-H64-I64	=$F$28	=IF((K64-J64)/(144*E64)>0,(K64-J64)/(144*E64),0)	=C64-14.7	=VLOOKUP(A64,$O$122:$K$153,2)
65	160302	167.8	18.07	5.696	0.016438	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B65<173,$F$18*(173/B65),$F$18),2)	=((14.7-D65)*144*E65)+F65-G65-H65-I65	=$F$28	=IF((K65-J65)/(144*E65)>0,(K65-J65)/(144*E65),0)	=C65-14.7	=VLOOKUP(A65,$O$122:$K$153,2)
66	170302	166.6	17.9	5.539	0.016431	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B66<173,$F$18*(173/B66),$F$18),2)	=((14.7-D66)*144*E66)+F66-G66-H66-I66	=$F$28	=IF((K66-J66)/(144*E66)>0,(K66-J66)/(144*E66),0)	=C66-14.7	=VLOOKUP(A66,$O$122:$K$153,2)
67	180302	165.3	17.72	5.374	0.016424	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B67<173,$F$18*(173/B67),$F$18),2)	=((14.7-D67)*144*E67)+F67-G67-H67-I67	=$F$28	=IF((K67-J67)/(144*E67)>0,(K67-J67)/(144*E67),0)	=C67-14.7	=VLOOKUP(A67,$O$122:$K$153,2)
68	180302	164.1	17.54	5.225	0.016417	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B68<173,$F$18*(173/B68),$F$18),2)	=((14.7-D68)*144*E68)+F68-G68-H68-I68	=$F$28	=IF((K68-J68)/(144*E68)>0,(K68-J68)/(144*E68),0)	=C68-14.7	=VLOOKUP(A68,$O$122:$K$153,2)
69	184052	163.8	17.47	5.184	0.016414	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B69<173,$F$18*(173/B69),$F$18),2)	=((14.7-D69)*144*E69)+F69-G69-H69-I69	=$F$28	=IF((K69-J69)/(144*E69)>0,(K69-J69)/(144*E69),0)	=C69-14.7	=VLOOKUP(A69,$O$122:$K$153,2)
70	186162	163.3	17.43	5.127	0.016413	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B70<173,$F$18*(173/B70),$F$18),2)	=((14.7-D70)*144*E70)+F70-G70-H70-I70	=$F$28	=IF((K70-J70)/(144*E70)>0,(K70-J70)/(144*E70),0)	=C70-14.7	=VLOOKUP(A70,$O$122:$K$153,2)
71	197808	163.2	17.41	5.115	0.016412	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B71<173,$F$18*(173/B71),$F$18),2)	=((14.7-D71)*144*E71)+F71-G71-H71-I71	=$F$28	=IF((K71-J71)/(144*E71)>0,(K71-J71)/(144*E71),0)	=C71-14.7	=VLOOKUP(A71,$O$122:$K$153,2)
72	200302	162.9	17.37	5.079	0.016411	=$F$25	=0.00000025*$F$9^2	=0.38*(($F$9/4000)^2)	=ROUND(IF(B72<173,$F$18*(173/B72),$F$18),2)	=((14.7-D72)*144*E72)+F72-G72-H72-I72	=$F$28	=IF((K72-J72)/(144*E72)>0,(K72-J72)/(144*E72),0)	=C72-14.7	=VLOOKUP(A72,$O$122:$K$153,2)

RHR - Long Term (After EPU) 1.5 wt. % Containment Leakage & 100% Spray Efficiency

	A	B	C	D	E	F	G	H	I	J	K	L	M	N
75-77	Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pv (psia)	vf (ft^3/lb)	Z (ft)	hf (ft)	hw (ft)	hd (ft)	Avail NPSHa (ft)	Req NPSHr (ft)	Req COP (psig)	CPA (psig)	CPC (psig)
78	795.6343	168.7	17.71	5.851	0.016449	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B78<173,$C$18*(173/B78),$C$18),2)	=((14.7-D78)*144*E78)+F78-G78+H78-I78	=$C$26	=IF(((K78-J78)/(144*E78))>0,((K78-J78)/(144*E78)),0)	=C78-14.7	=VLOOKUP(A78,$G$122:$K$153,2)
79	1097.893	171.8	17.94	6.246	0.016461	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B79<173,$C$18*(173/B79),$C$18),2)	=((14.7-D79)*144*E79)+F79-G79+H79-I79	=$C$26	=IF(((K79-J79)/(144*E79))>0,((K79-J79)/(144*E79)),0)	=C79-14.7	=VLOOKUP(A79,$G$122:$K$153,2)
80	2033.355	175.8	18.57	6.862	0.016480	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B80<173,$C$18*(173/B80),$C$18),2)	=((14.7-D80)*144*E80)+F80-G80+H80-I80	=$C$26	=IF(((K80-J80)/(144*E80))>0,((K80-J80)/(144*E80)),0)	=C80-14.7	=VLOOKUP(A80,$G$122:$K$153,2)
81	2962.323	180	19.17	7.511	0.016500	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B81<173,$C$18*(173/B81),$C$18),2)	=((14.7-D81)*144*E81)+F81-G81+H81-I81	=$C$26	=IF(((K81-J81)/(144*E81))>0,((K81-J81)/(144*E81)),0)	=C81-14.7	=VLOOKUP(A81,$G$122:$K$153,2)
82	4198.48	183.4	19.9	8.086	0.01652	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B82<173,$C$18*(173/B82),$C$18),2)	=((14.7-D82)*144*E82)+F82-G82+H82-I82	=$C$26	=IF(((K82-J82)/(144*E82))>0,((K82-J82)/(144*E82)),0)	=C82-14.7	=VLOOKUP(A82,$G$122:$K$153,2)
83	5125.043	186.2	20.34	8.42	0.016541	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B83<173,$C$18*(173/B83),$C$18),2)	=((14.7-D83)*144*E83)+F83-G83+H83-I83	=$C$26	=IF(((K83-J83)/(144*E83))>0,((K83-J83)/(144*E83)),0)	=C83-14.7	=VLOOKUP(A83,$G$122:$K$153,2)
84	6275.292	187	20.82	8.796	0.016552	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B84<173,$C$18*(173/B84),$C$18),2)	=((14.7-D84)*144*E84)+F84-G84+H84-I84	=$C$26	=IF(((K84-J84)/(144*E84))>0,((K84-J84)/(144*E84)),0)	=C84-14.7	=VLOOKUP(A84,$G$122:$K$153,2)
85	8026.636	189.1	21.5	9.181	0.016566	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B85<173,$C$18*(173/B85),$C$18),2)	=((14.7-D85)*144*E85)+F85-G85+H85-I85	=$C$26	=IF(((K85-J85)/(144*E85))>0,((K85-J85)/(144*E85)),0)	=C85-14.7	=VLOOKUP(A85,$G$122:$K$153,2)
86	10219.80	191	21.80	9.541	0.016576	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B86<173,$C$18*(173/B86),$C$18),2)	=((14.7-D86)*144*E86)+F86-G86+H86-I86	=$C$26	=IF(((K86-J86)/(144*E86))>0,((K86-J86)/(144*E86)),0)	=C86-14.7	=VLOOKUP(A86,$G$122:$K$153,2)
87	12083.67	192.2	22.06	9.780	0.016588	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B87<173,$C$18*(173/B87),$C$18),2)	=((14.7-D87)*144*E87)+F87-G87+H87-I87	=$C$26	=IF(((K87-J87)/(144*E87))>0,((K87-J87)/(144*E87)),0)	=C87-14.7	=VLOOKUP(A87,$G$122:$K$153,2)
88	15189.8	193.8	22.31	10.083	0.016594	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B88<173,$C$18*(173/B88),$C$18),2)	=((14.7-D88)*144*E88)+F88-G88+H88-I88	=$C$26	=IF(((K88-J88)/(144*E88))>0,((K88-J88)/(144*E88)),0)	=C88-14.7	=VLOOKUP(A88,$G$122:$K$153,2)
89	17668.2	194.3	22.43	10.233	0.016598	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B89<173,$C$18*(173/B89),$C$18),2)	=((14.7-D89)*144*E89)+F89-G89+H89-I89	=$C$26	=IF(((K89-J89)/(144*E89))>0,((K89-J89)/(144*E89)),0)	=C89-14.7	=VLOOKUP(A89,$G$122:$K$153,2)
90	20156.36	194.6	22.46	10.298	0.016601	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B90<173,$C$18*(173/B90),$C$18),2)	=((14.7-D90)*144*E90)+F90-G90+H90-I90	=$C$26	=IF(((K90-J90)/(144*E90))>0,((K90-J90)/(144*E90)),0)	=C90-14.7	=VLOOKUP(A90,$G$122:$K$153,2)
91	23811.7	194.7	22.48	10.32	0.016601	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B91<173,$C$18*(173/B91),$C$18),2)	=((14.7-D91)*144*E91)+F91-G91+H91-I91	=$C$26	=IF(((K91-J91)/(144*E91))>0,((K91-J91)/(144*E91)),0)	=C91-14.7	=VLOOKUP(A91,$G$122:$K$153,2)
92	24494.98	194.7	22.48	10.32	0.016601	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B92<173,$C$18*(173/B92),$C$18),2)	=((14.7-D92)*144*E92)+F92-G92+H92-I92	=$C$26	=IF(((K92-J92)/(144*E92))>0,((K92-J92)/(144*E92)),0)	=C92-14.7	=VLOOKUP(A92,$G$122:$K$153,2)
93	25110.98	194.7	22.47	10.32	0.016601	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B93<173,$C$18*(173/B93),$C$18),2)	=((14.7-D93)*144*E93)+F93-G93+H93-I93	=$C$26	=IF(((K93-J93)/(144*E93))>0,((K93-J93)/(144*E93)),0)	=C93-14.7	=VLOOKUP(A93,$G$122:$K$153,2)
94	30084.64	194.3	22.42	10.233	0.016599	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B94<173,$C$18*(173/B94),$C$18),2)	=((14.7-D94)*144*E94)+F94-G94+H94-I94	=$C$26	=IF(((K94-J94)/(144*E94))>0,((K94-J94)/(144*E94)),0)	=C94-14.7	=VLOOKUP(A94,$G$122:$K$153,2)
95	35084.51	193.7	22.33	10.104	0.016596	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B95<173,$C$18*(173/B95),$C$18),2)	=((14.7-D95)*144*E95)+F95-G95+H95-I95	=$C$26	=IF(((K95-J95)/(144*E95))>0,((K95-J95)/(144*E95)),0)	=C95-14.7	=VLOOKUP(A95,$G$122:$K$153,2)
96	40020.36	192.8	22.2	9.914	0.016590	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B96<173,$C$18*(173/B96),$C$18),2)	=((14.7-D96)*144*E96)+F96-G96+H96-I96	=$C$26	=IF(((K96-J96)/(144*E96))>0,((K96-J96)/(144*E96)),0)	=C96-14.7	=VLOOKUP(A96,$G$122:$K$153,2)
97	45437.23	191.5	22.01	9.644	0.016581	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B97<173,$C$18*(173/B97),$C$18),2)	=((14.7-D97)*144*E97)+F97-G97+H97-I97	=$C$26	=IF(((K97-J97)/(144*E97))>0,((K97-J97)/(144*E97)),0)	=C97-14.7	=VLOOKUP(A97,$G$122:$K$153,2)
98	49406.11	190.4	21.78	9.42	0.016574	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B98<173,$C$18*(173/B98),$C$18),2)	=((14.7-D98)*144*E98)+F98-G98+H98-I98	=$C$26	=IF(((K98-J98)/(144*E98))>0,((K98-J98)/(144*E98)),0)	=C98-14.7	=VLOOKUP(A98,$G$122:$K$153,2)
99	60550.64	187.3	21.21	8.794	0.016554	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B99<173,$C$18*(173/B99),$C$18),2)	=((14.7-D99)*144*E99)+F99-G99+H99-I99	=$C$26	=IF(((K99-J99)/(144*E99))>0,((K99-J99)/(144*E99)),0)	=C99-14.7	=VLOOKUP(A99,$G$122:$K$153,2)
100	70342.29	184.4	20.72	8.276	0.016536	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B100<173,$C$18*(173/B100),$C$18),2)	=((14.7-D100)*144*E100)+F100-G100+H100-I100	=$C$26	=IF(((K100-J100)/(144*E100))>0,((K100-J100)/(144*E100)),0)	=C100-14.7	=VLOOKUP(A100,$G$122:$K$153,2)
101	80342.29	181.8	20.20	7.816	0.01652	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B101<173,$C$18*(173/B101),$C$18),2)	=((14.7-D101)*144*E101)+F101-G101+H101-I101	=$C$26	=IF(((K101-J101)/(144*E101))>0,((K101-J101)/(144*E101)),0)	=C101-14.7	=VLOOKUP(A101,$G$122:$K$153,2)
102	90330.91	179.3	19.86	7.395	0.016506	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B102<173,$C$18*(173/B102),$C$18),2)	=((14.7-D102)*144*E102)+F102-G102+H102-I102	=$C$26	=IF(((K102-J102)/(144*E102))>0,((K102-J102)/(144*E102)),0)	=C102-14.7	=VLOOKUP(A102,$G$122:$K$153,2)
103	100330.8	176.8	19.52	6.994	0.01649	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B103<173,$C$18*(173/B103),$C$18),2)	=((14.7-D103)*144*E103)+F103-G103+H103-I103	=$C$26	=IF(((K103-J103)/(144*E103))>0,((K103-J103)/(144*E103)),0)	=C103-14.7	=VLOOKUP(A103,$G$122:$K$153,2)
104	110330.7	174.8	19.2	6.686	0.016478	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B104<173,$C$18*(173/B104),$C$18),2)	=((14.7-D104)*144*E104)+F104-G104+H104-I104	=$C$26	=IF(((K104-J104)/(144*E104))>0,((K104-J104)/(144*E104)),0)	=C104-14.7	=VLOOKUP(A104,$G$122:$K$153,2)
105	120306.6	173.2	18.93	6.447	0.016469	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B105<173,$C$18*(173/B105),$C$18),2)	=((14.7-D105)*144*E105)+F105-G105+H105-I105	=$C$26	=IF(((K105-J105)/(144*E105))>0,((K105-J105)/(144*E105)),0)	=C105-14.7	=VLOOKUP(A105,$G$122:$K$153,2)
106	130302	171.8	18.69	6.245	0.016461	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B106<173,$C$18*(173/B106),$C$18),2)	=((14.7-D106)*144*E106)+F106-G106+H106-I106	=$C$26	=IF(((K106-J106)/(144*E106))>0,((K106-J106)/(144*E106)),0)	=C106-14.7	=VLOOKUP(A106,$G$122:$K$153,2)
107	140302	170.4	18.47	6.048	0.016453	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B107<173,$C$18*(173/B107),$C$18),2)	=((14.7-D107)*144*E107)+F107-G107+H107-I107	=$C$26	=IF(((K107-J107)/(144*E107))>0,((K107-J107)/(144*E107)),0)	=C107-14.7	=VLOOKUP(A107,$G$122:$K$153,2)
108	150302	169.1	18.27	5.87	0.016445	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B108<173,$C$18*(173/B108),$C$18),2)	=((14.7-D108)*144*E108)+F108-G108+H108-I108	=$C$26	=IF(((K108-J108)/(144*E108))>0,((K108-J108)/(144*E108)),0)	=C108-14.7	=VLOOKUP(A108,$G$122:$K$153,2)
109	160302	167.8	18.07	5.696	0.016438	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B109<173,$C$18*(173/B109),$C$18),2)	=((14.7-D109)*144*E109)+F109-G109+H109-I109	=$C$26	=IF(((K109-J109)/(144*E109))>0,((K109-J109)/(144*E109)),0)	=C109-14.7	=VLOOKUP(A109,$G$122:$K$153,2)
110	170302	166.6	17.9	5.538	0.016431	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B110<173,$C$18*(173/B110),$C$18),2)	=((14.7-D110)*144*E110)+F110-G110+H110-I110	=$C$26	=IF(((K110-J110)/(144*E110))>0,((K110-J110)/(144*E110)),0)	=C110-14.7	=VLOOKUP(A110,$G$122:$K$153,2)
111	180302	165.3	17.72	5.374	0.016424	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B111<173,$C$18*(173/B111),$C$18),2)	=((14.7-D111)*144*E111)+F111-G111+H111-I111	=$C$26	=IF(((K111-J111)/(144*E111))>0,((K111-J111)/(144*E111)),0)	=C111-14.7	=VLOOKUP(A111,$G$122:$K$153,2)
112	190302	164.1	17.54	5.225	0.016417	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B112<173,$C$18*(173/B112),$C$18),2)	=((14.7-D112)*144*E112)+F112-G112+H112-I112	=$C$26	=IF(((K112-J112)/(144*E112))>0,((K112-J112)/(144*E112)),0)	=C112-14.7	=VLOOKUP(A112,$G$122:$K$153,2)
113	194052	163.6	17.47	5.164	0.016414	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B113<173,$C$18*(173/B113),$C$18),2)	=((14.7-D113)*144*E113)+F113-G113+H113-I113	=$C$26	=IF(((K113-J113)/(144*E113))>0,((K113-J113)/(144*E113)),0)	=C113-14.7	=VLOOKUP(A113,$G$122:$K$153,2)
114	196552	163.3	17.43	5.127	0.016413	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B114<173,$C$18*(173/B114),$C$18),2)	=((14.7-D114)*144*E114)+F114-G114+H114-I114	=$C$26	=IF(((K114-J114)/(144*E114))>0,((K114-J114)/(144*E114)),0)	=C114-14.7	=VLOOKUP(A114,$G$122:$K$153,2)
115	197802	163.2	17.41	5.116	0.016412	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B115<173,$C$18*(173/B115),$C$18),2)	=((14.7-D115)*144*E115)+F115-G115+H115-I115	=$C$26	=IF(((K115-J115)/(144*E115))>0,((K115-J115)/(144*E115)),0)	=C115-14.7	=VLOOKUP(A115,$G$122:$K$153,2)
116	200302	162.9	17.37	5.079	0.016411	=$C$25	=0.0000000477*$C$39^2	=$C$15	=ROUND(IF(B116<173,$C$18*(173/B116),$C$18),2)	=((14.7-D116)*144*E116)+F116-G116+H116-I116	=$C$26	=IF(((K116-J116)/(144*E116))>0,((K116-J116)/(144*E116)),0)	=C116-14.7	=VLOOKUP(A116,$G$122:$K$153,2)

VY CALCULATION DATABASE INPUT FORM

Attachment B Page 1 of 2

Place this form in the calculation package immediately following the Title page or CCN form.

VY Calculation/CCN Number: VYC-0808 / 06
Revision Number: 6
Vendor Calculation Number: N/A
Revision Number: N/A
Vendor Name: N/A
PO Number: N/A
Originating Department: Fluid Systems
Critical References Impacted: ☒ UFSAR ☒ DBD ☐ Reload. "Check" the appropriate box if any critical document is identified in the tables below.
EMPAC Asset/Equipment ID Number(s): P-10-1A/B/C/D, P-46-1A/B
EMPAC Asset/System ID Number(s): 10, 14
Keywords: Residual Heat Removal (RHR), Core Spray (CS), Net Positive Suction Head (NPSH), Loss of Coolant Accident (LOCA), Extended Power Uprate (EPU)

For Revision/CCN only: Are deletions to General References, Design Input Documents or Design Output Documents required? ☐ Yes† ☒ No

Design Input Documents and General References - The following documents provide design input or supporting information to this calculation. (Refer to Appendix A, sections 3.2.7 and section 4)

* Reference #	** DOC #	REV #	***Document Title (including Date, if applicable)	Significant Difference Review ††	**** Affected Program	Critical Reference (✓)
1	GE-VYNPS-AEP-346	1	VYNPS EPU T0400: DBA-LOCA for Long Term NPSH evaluation (7/6/04)			
2	VYC-1924	0	DE&S Calc. DC-A34600-006. Vermont Yankee ECCS Suction Strainer Head Loss Performance Assessment, RHR and CS Debris Head Loss Calculations			
3	VYC-0808 / CCN 03	6	Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident			
4	VYC-0808	6	Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident			
5	ERC-2004-024	N/A	Revised OPL-4 Input for Minimum DW Pressure Case dated 6/8/04			
6			NOT USED			
7			ASME Steam Tables, 1967 IFC Formulation for Industrial Use			
8	ERC No. 2003-027	N/A	Debris Source Terms Appropriate for Power Uprate Evaluations of ECCS NPSH (5/13/03)			
9	RHR	1/IC16	Design Basis Document for Residual Heat Removal System			✓
10	VYC-1924 / CCN 02	0	DE&S Calc. DC-A34600-006. Vermont Yankee ECCS Suction Strainer Head Loss Performance Assessment, RHR and CS Debris Head Loss Calculations			
11	VYC-0808 / CCN 04	6	Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident or Anticipated Transients Without Scram			
12	VYC-0808 / CCN 05	6	Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident or Anticipated Transients Without Scram			

Design Output Documents - This calculation provides output to the following documents. (Refer to Appendix A, section 5)

* Reference #	** DOC #	REV #	Document Title (including Date, if applicable)	**** Affected Program	†††Critical Reference (✓)
12	NEDC-33090P	0	Safety Analysis Report For VYNPS Constant Pressure Power Uprate (PUSAR)		
13	VYC-1628	0	Torus Temperature and Pressure Response to Large Break LOCA and MSLB Accident Scenarios (may be superseded be GE EPU analysis)		
14	SADBD	2/IC2	Topical Design Basis Document for Safety Analysis		✓
15	UFSAR	17	Updated Final Safety Analysis Report		✓
16	RHR	1/IC16	Design Basis Document for Residual Heat Removal System		✓
17	CS	0/IC10	Design Basis Document for Core Spray System		✓

* Reference # - Assigned by preparer to identify the reference in the body of the calculation.

** Doc # - Identifying number on the document, if any (e.g., 5920-0264, G191172, VYC-1286)

*** Document Title - List the specific documentation in this column. "See attached list" is not acceptable. Design Input/Output Documents should identify the specific design input document used in the calculation or the specific document affected by the calculation and not simply reference the document (e.g., VYDC, MM) that the calculation was written to support. If a DBD is used as a general reference, include the most current interim change number after the title.

**** Affected Program - List the affected program or the program that reference is related to or part of.

† If "yes," attach a copy of "VY Calculation Data" marked-up to reflect deletion (See Section 3.1.8 for Revision and 5.2.3.18 for CCNs).

†† If the listed input is a calculation listed in the calculation database that is not a calculation of record (see definition), place a check mark in this space to indicate completion of the required significant difference review. (see Appendix A, section 4.1.4.4.3). Otherwise, enter "N/A."

††† If the reference is UFSAR, DBD or Reload (IASD or OPL), check Critical Reference column and check UFSAR, DBD or Reload, as appropriate, on this form (above).

U.S. NUCLEAR REGULATORY COMMISSION July 2000

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.183
(Draft was issued as DG-1081)

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

Docket No. 50-271
DPS Exhibit #22
31 Pages

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APPENDICES

A. INTRODUCTION

This guide provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.34, "Contents of Applications; Technical Information," requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. In 10 CFR Part 100, "Reactor Site Criteria," Section 100.11,[1] "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Ref. 1), is cited in 10 CFR Part 100 as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification of equipment under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, "Clarification of TMI Action Plan Requirements" (Ref. 2). The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility final safety analysis report (FSAR). Fundamental assumptions that are design inputs, including the source term, are to be included in the FSAR and become part of the facility design basis.[2]

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January

[1] Applicants for a construction permit, a design certification, or a combined license that do not reference a standard design certification who applied after January 10, 1997, are required by regulation to meet radiological criteria provided in 10 CFR 50.34.

[2] As defined in 10 CFR 50.2, *design bases* means information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculation or experiments or both) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. The NRC considers the accident source term to be an integral part of the design basis because it sets forth specific values (or a range of values) for controlling parameters that constitute reference bounds for design.

10, 1997, is allowed by 10 CFR 50.67, "Accident Source Term," to voluntarily revise the accident source term used in design basis radiological consequence analyses.

In general, information provided by regulatory guides is reflected in NUREG-0800, the Standard Review Plan (SRP) (Ref 3). The NRC staff uses the SRP to review applications to construct and operate nuclear power plants. This regulatory guide applies to Chapter 15.0.1 of the SRP.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-0011. The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

B. DISCUSSION

An accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA). Although the LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. The design basis accidents (DBAs) were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired depth is achieved, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The NRC's policy statement on the use of PRA methods (Ref. 4) calls for the use of PRA technology in all regulatory matters in a manner that complements the NRC's deterministic approach and supports the traditional defense-in-depth philosophy.

Since the publication of TID-14844 (Ref. 1), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 5). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an AST in analyses to support cost-beneficial licensing actions.

The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST[3] in design basis analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and this regulatory guide.

The NRC's traditional methods for calculating the radiological consequences of design basis accidents are described in a series of regulatory guides and SRP chapters. That guidance was developed to be consistent with the TID-14844 source term and the whole body and thyroid dose guidelines stated in 10 CFR 100.11. Many of those analysis assumptions and methods are inconsistent with the ASTs and with the total effective dose equivalent (TEDE) criteria provided in 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST. This guidance supersedes corresponding radiological analysis assumptions provided in other regulatory guides and SRP chapters when used in conjunction with an approved AST and the TEDE criteria provided in 10 CFR 50.67. The affected guides will not be withdrawn as their guidance still applies when an AST is not used. Specifically, the affected regulatory guides are:

Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" (Ref. 6)

Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" (Ref. 7)

Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors" (Ref. 8)

Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Ref. 9)

Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" (Ref. 10)

The guidance in Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant." (Ref. 11), regarding the radiological source term used in the determination of integrated doses for environmental qualification purposes is superseded by the corresponding guidance in this regulatory guide for those facilities that are proposing to, or have already, implemented an AST. All other guidance in Regulatory Guide 1.89 remains effective.

This guide primarily addresses design basis accidents, such as those addressed in Chapter 15 of typical final safety analysis reports (FSARs). This guide does not address all areas of potentially significant risk. Although this guide addresses fuel handling accidents, other events that could occur during shutdown operations are not currently addressed. The NRC staff has several ongoing

[3] The NUREG-1465 source terms have often been referred to as the "revised source terms." In recognition that there may be additional source terms identified in the future, 10 CFR 50.67 addresses "alternative source terms." This regulatory guide endorses a source term derived from NUREG-1465 and provides guidance on the acceptable attributes of other alternative source terms.

initiatives involving risks of shutdown operations, extended burnup fuels, and risk-informing current regulations. The information in this guide may be revised in the future as NRC staff evaluations are completed and regulatory decisions on these issues are made.

C. REGULATORY POSITION

1. IMPLEMENTATION OF AST

1.1 Generic Considerations

As used in this guide, an AST is an accident source term that is different from the accident source term used in the original design and licensing of the facility and that has been approved for use under 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and identifies significant characteristics of other ASTs that may be found acceptable. While the NRC staff recognizes several potential uses of an AST, it is not possible to foresee all possible uses. The NRC staff will allow licensees to pursue technically justifiable uses of the ASTs in the most flexible manner compatible with maintaining a clear, logical, and consistent design basis. The NRC staff will approve these license amendment requests if the facility, as modified, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs.

1.1.1 Safety Margins

The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the initial AST implementation has been approved by the staff and has become part of the facility design basis, the licensee may use 10 CFR 50.59 and its supporting guidance in assessing safety margins related to subsequent facility modifications and changes to procedures.

1.1.2 Defense in Depth

The proposed uses of an AST and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.

Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.

1.1.3 Integrity of Facility Design Basis

The design basis accident source term is a fundamental assumption upon which a significant portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. Although a complete re-assessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses would generally not be necessary. Regulatory Position 1.3 of this guide provides guidance on which analyses need updating as part of the AST implementation submittal and which may need updating in the future as additional modifications are performed.

This approach would create two tiers of analyses, those based on the previous source term and those based on an AST. The radiological acceptance criteria would also be different with some analyses based on whole body and thyroid criteria and some based on TEDE criteria. Full implementation of the AST revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. Selective implementation of the AST also revises the plant licensing basis and may establish the TEDE dose as the new acceptance criteria. Selective implementation differs from full implementation only in the scope of the change. In either case, the facility design bases should clearly indicate that the source term assumptions and radiological criteria in these affected analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria.

Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses, radiological and nonradiological, unless these data would result in nonconservative results or otherwise conflict with the guidance in this guide.

1.1.4 Emergency Preparedness Applications

Requirements for emergency preparedness at nuclear power plants are set forth in 10 CFR 50.47, "Emergency Plans." Additional requirements are set forth in Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50. The planning basis for many of these requirements was published in NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants"[4] (Ref. 12). This joint effort by the Environmental Protection Agency (EPA) and the NRC considered the principal characteristics (such as nuclides released and distances) likely to be involved for a spectrum of design basis and severe (core melt) accidents. No single accident scenario is the basis of the required preparedness. The objective of the planning is to provide public protection that would encompass a wide spectrum of possible events with a sufficient basis for extension of response efforts for unanticipated events. These requirements were issued after a long period of involvement by numerous stakeholders, including the Federal Emergency Management Agency, other Federal agencies, local and State governments (and in some cases, foreign governments), private citizens, utilities, and industry groups.

Although the AST provided in this guide was based on a limited spectrum of severe accidents, the particular characteristics have been tailored specifically for DBA analysis use. The AST is not

[4] This planning basis is also addressed in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (Ref. 13).

representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient *by itself* as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.

This guidance does not, however, preclude the appropriate use of the insights of the AST in establishing emergency response procedures such as those associated with emergency dose projections, protective measures, and severe accident management guides.

1.2 Scope of Implementation

The AST described in this guide is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. The accident source term is a fundamental assumption upon which a large portion of the facility design is based. Additionally, many aspects of facility operation derive from the design analyses that incorporated the earlier accident source term. A complete implementation of an AST would upgrade all existing radiological analyses and would consider the impact of all five characteristics of the AST as defined in 10 CFR 50.2. However, the NRC staff has determined that there could be implementations for which this level of re-analysis may not be necessary. Two categories are defined: Full and selective implementations.

1.2.1 Full Implementation

Full implementation is a modification of the facility design basis that addresses all characteristics of the AST, that is, composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses. At a minimum for full implementations, the DBA LOCA must be re-analyzed using the guidance in Appendix A of this guide. Additional guidance on analysis is provided in Regulatory Position 1.3 of this guide. Since the AST and TEDE criteria would become part of the facility design basis, new applications of the AST would not require prior NRC approval unless stipulated by 10 CFR 50.59, "Changes, Tests, and Experiments," or unless the new application involved a change to a technical specification. However, a change from an approved AST to a different AST that is not approved for use at that facility would require a license amendment under 10 CFR 50.67.

1.2.2 Selective Implementation

Selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails re-evaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees flexibility in technically justified selective implementations provided a clear, logical, and consistent design basis is maintained. An example of an application of selective implementation would be one in which a licensee desires to use the release timing insights of the AST to increase the required closure time for a containment isolation valve by a small amount. Another example would be a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. For the latter, the licensee may only need to re-analyze DBAs that credited the iodine removal by the charcoal media. Additional analysis guidance is provided in Regulatory Position 1.3 of this guide. NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. The

licensee would be able to make subsequent modifications to the facility and changes to procedures based on the selected AST characteristics incorporated into the design basis under the provisions of 10 CFR 50.59. However, use of other characteristics of an AST or use of TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, would require prior staff approval under 10 CFR 50.67. As an example, a licensee with an implementation involving only timing, such as relaxed closure time on isolation valves, could not use 10 CFR 50.59 as a mechanism to implement a modification involving a reanalysis of the DBA LOCA. However, this licensee could extend use of the timing characteristic to adjust the closure time on isolation valves not included in the original approval.

1.3 Scope of Required Analyses

1.3.1 Design Basis Radiological Analyses

There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. These requirements include, but are not limited to, the following.

- Environmental Qualification of Equipment (10 CFR 50.49)
- Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50)
- Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)
- Alternative Source Term (10 CFR 50.67)
- Environmental Reports (10 CFR Part 51)
- Facility Siting (10 CFR 100.11)[5]

There may be additional applications of the accident source term identified in the technical specification bases and in various licensee commitments. These include, but are not limited to, the following from Reference 2, NUREG-0737.

- Post-Accident Access Shielding (NUREG-0737, II.B.2)
- Post-Accident Sampling Capability (NUREG-0737, II.B.3)
- Accident Monitoring Instrumentation (NUREG-0737, II.F.1)
- Leakage Control (NUREG-0737, III.D.1.1)
- Emergency Response Facilities (NUREG-0737, III.A.1.2)
- Control Room Habitability (NUREG-0737, III.D.3.4)

1.3.2 Re-Analysis Guidance

Any implementation of an AST, full or selective, and any associated facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-

[5] Dose guidelines of 10 CFR 100.11 are superseded by 10 CFR 50.67 for licensees that have implemented an AST.

evaluation will necessarily be a function of the specific proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. The license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

The NRC staff has performed an evaluation of the impact of the AST on three representative operating reactors (Ref. 14). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions (Ref. 1) and the whole body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt the licensee from evaluating the remaining radiological and nonradiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application might be acceptable without dose calculations. However, evaluations may need to be performed regarding the ability of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.

For full implementation, a complete DBA LOCA analysis as described in Appendix A of this guide should be performed, as a minimum. Other design basis analyses are updated in accordance with the guidance in this section.

A selective implementation of an AST and any associated facility modification based on the AST should evaluate all the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. Design basis analyses are updated in accordance with the guidance in this section. There is no minimum requirement that a DBA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected characteristics of the AST, and if dose calculations are performed, the TEDE criteria. For selective implementations based on the timing characteristic of the AST, e.g., change in the closure timing of a containment isolation valve, re-analysis of radiological calculations may not be

[6] For example, a proposed modification to change the timing of a containment isolation valve from 2.5 seconds to 5.0 seconds might be acceptable without any dose calculations. However, a proposed modification that would delay containment spray actuation could involve recalculation of DBA LOCA doses, re-assessment of the containment pressure and temperature transient, recalculation of sump pH, re-assessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.

necessary if the modified elapsed time remains a fraction (e.g., 25%) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. If affected design basis analyses are to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed.

1.3.3 Use of Sensitivity or Scoping Analyses

It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a *sensitivity analysis* is an evaluation that considers how the overall results vary as an input parameter (in this case, AST characteristics) is varied. A *scoping analysis* is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include postaccident vital area access dose calculations, shielding calculations, and equipment environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze that case, and use the results to draw conclusions regarding the remainder of the analyses. It may also be possible to show that for some analyses the whole body and thyroid doses determined with the previous source term would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or control room dose.

1.3.4 Updating Analyses Following Implementation

Full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. The implementation may have been supported in part by sensitivity or scoping analyses that concluded many of the design basis radiological analyses would remain bounding for the AST and the TEDE criteria and would not require updating. After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses on an individual as-needed basis. Re-evaluation using the previously approved source term may not be appropriate. Since the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility do not constitute a change in analysis methodology that would require NRC approval.[7]

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[7] In performing screenings and evaluations pursuant to 10 CFR 50.59, it may be necessary to compare dose results expressed in terms of whole body and thyroid with new results expressed in terms of TEDE. In these cases, the previous thyroid dose should be multiplied by 0.03 and the product added to the whole body dose. The result is then compared to the TEDE result in the screenings and evaluations. This change in dose methodology is not considered a change in the method of evaluation if the licensee was previously authorized to use an AST and the TEDE criteria under 10 CFR 50.67.

This guidance is also applicable to selective implementations to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the characteristics of the AST and TEDE criteria identified in the facility design basis need to be considered in updating the analyses. Use of other characteristics of the AST or TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics, requires prior NRC staff approval under 10 CFR 50.67.

1.3.5 Equipment Environmental Qualification

Current environmental qualification (EQ) analyses may be impacted by a proposed plant modification associated with the AST implementation. The EQ analyses that have assumptions or inputs affected by the plant modification should be updated to address these impacts. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue. The EQ dose estimates should be calculated using the design basis survivability period.

1.4 Risk Implications

The use of an AST changes only the regulatory assumptions regarding the analytical treatment of the design basis accidents. The AST has no direct effect on the probability of the accident. Use of an AST alone cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the impact on the existing PRAs should be evaluated.

Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety.

The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. For guidance, refer to Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 15).

1.5 Submittal Requirements

According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form prescribed for original applications. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)" (Ref 16), provides additional guidance. The NRC staff's finding that the amendment may be approved must be based on the licensee's analyses,

since it is these analyses that will become part of the design basis of the facility. The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.

If the licensee has used a current approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.

1.6 FSAR Requirements

Requirements for updating the facility's final safety analysis report (FSAR) are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include all changes made in the facility or procedures described in the FSAR and all safety evaluations performed by the licensee in support of requests for license amendments or in support of conclusions that changes did not involve unreviewed safety questions. The analyses required by 10 CFR 50.67 are subject to this requirement. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis source term by the AST. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 16) provides additional guidance. The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.

2. ATTRIBUTES OF AN ACCEPTABLE AST

An acceptable AST is not set forth in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an AST that is acceptable to the NRC staff for use at operating power reactors. A substantial effort was expended by the NRC, its contractors, various national laboratories, peer reviewers, and others in performing severe accident research and in developing the source terms provided in NUREG-1465 (Ref. 5). However, future research may identify opportunities for changes in these source terms. The NRC staff will consider applications for an AST different from that identified in this guide. However, the NRC staff does not expect to approve any source term that is not of the same level of quality as the source terms in NUREG-1465. To be considered acceptable, an AST must have the following attributes:

2.1 The AST must be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events, that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.

2.2 The AST must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

2.3 The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

2.4 The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse.

2.5 The AST must be peer-reviewed by appropriately qualified subject matter experts. The peer-review comments and their resolution should be part of the documentation supporting the AST.

3. ACCIDENT SOURCE TERM

This section provides an AST that is acceptable to the NRC staff. The data in Regulatory Positions 3.2 through 3.5 are fundamental to the definition of an AST. Once approved, the AST assumptions or parameters specified in these positions become part of the facility's design basis. Deviations from this guidance must be evaluated against Regulatory Position 2. After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67.

3.1 Fission Product Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[8] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.

[8] The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02.

[9] Note that for some radionuclides, such as Cs-137, equilibrium will not be reached prior to fuel offload. Thus, the maximum inventory at the end of life should be used.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.

No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions[10]

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

Table 1
BWR Core Inventory Fraction Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.20	0.25
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

[10] The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.

Table 2
PWR Core Inventory Fraction
Released Into Containment

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali Metals	0.05	0.25	0.3
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

Table 3[11]
Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

3.3 Timing of Release Phases

Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[12] For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

[11] The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.

[12] In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase, i.e., in step increases.

Table 4
LOCA Release Phases

Phase	**PWRs Onset**	**PWRs Duration**	**BWRs Onset**	**BWRs Duration**
Gap Release	30 sec	0.5 hr	2 min	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr	0.5 hr	1.5 hr

For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.

3.4 Radionuclide Composition

Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.

Table 5
Radionuclide Groups

Group	**Elements**
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se, Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.

3.6 Fuel Damage in Non-LOCA DBAs

The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.

The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.

4. DOSE CALCULATIONAL METHODOLOGY

The NRC staff has determined that there is an implied synergy between the ASTs and total effective dose equivalent (TEDE) criteria, and between the TID-14844 source terms and the whole body and thyroid dose criteria, and therefore, they do not expect to allow the TEDE criteria to be used with TID-14844 calculated results. The guidance of this section applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67. Certain selective implementations may not require dose calculations as described in Regulatory Position 1.3 of this guide.

4.1 Offsite Dose Consequences

The following assumptions should be used in determining the TEDE for persons located at or beyond the boundary of the exclusion area (EAB):

4.1.1 The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.[13]

4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.

4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.

[13] The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.

4.1.4 The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.

4.1.5 The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67.[14] The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).

4.1.6 TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.

4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground.

4.2 Control Room Dose Consequences

The following guidance should be used in determining the TEDE for persons located in the control room:

4.2.1 The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,
- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,
- Radiation shine from the external radioactive plume released from the facility,

[14] With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.

- Radiation shine from radioactive material in the reactor containment,

- Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.

4.2.2 The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.

4.2.3 The models used to transport radioactive material into and through the control room,[15] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance. The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.

4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days.[16] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.

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[15] The iodine protection factor (IPF) methodology of Reference 22 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 23) and RADTRAD (Ref. 24) incorporate suitable methodologies.

[16] This occupancy is modeled in the χ/Q values determined in Reference 22 and should not be credited twice. The ARCON96 Code (Ref. 26) does not incorporate these occupancy assumptions, making it necessary to apply this correction in the dose calculations.

4.2.7 Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22).

$$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173} \qquad \text{Equation 1}$$

4.3 Other Dose Consequences

The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.

4.4 Acceptance Criteria

The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.

The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference General Design Criteria 19 (GDC 19) from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria are generally specified in terms of whole body dose, or its equivalent to any body organ. For facilities applying for, or having received, approval for the use of an AST, the applicable criteria should be updated for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

Table 6[17]
Accident Dose Criteria

Accident or Case	EAB and LPZ Dose Criteria	Analysis Release Duration
LOCA	25 rem TEDE	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steam Line Break		Instantaneous puff
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Equilibrium Iodine Activity	2.5 rem TEDE	
BWR Rod Drop Accident	6.3 rem TEDE	24 hours
PWR Steam Generator Tube Rupture		Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Main Steam Line Break		Until cold shutdown is established
Fuel Damage or Pre-incident Spike	25 rem TEDE	
Coincident Iodine Spike	2.5 rem TEDE	
PWR Locked Rotor Accident	2.5 rem TEDE	Until cold shutdown is established
PWR Rod Ejection Accident	6.3 rem TEDE	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel Handling Accident	6.3 rem TEDE	2 hours

The column labeled "Analysis Release Duration" is a summary of the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY

5.1 General Considerations

5.1.1 Analysis Quality

The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled

[17] For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steam line break analyses.

directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.

5.1.2 Credit for Engineered Safeguard Features

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[18] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.

5.1.4 Applicability of Prior Licensing Basis

The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific

[18] Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 25) and in Generic Letter 99-02 (Ref. 27) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address potential changes in the parameter between scheduled surveillance tests.

implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. This is not considered a backfit as defined by 10 CFR 50.109, "Backfitting." However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

5.2 Accident-Specific Assumptions

The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST.

The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing basis consideration. The assumptions in the appendices are deemed consistent with the AST identified in Regulatory Position 3 and internally consistent with each other. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect this consistency.

The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based upon risk insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.

5.3 Meteorology Assumptions

Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28).

References 22 and 28 should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96[19] (Ref. 26) is generally acceptable to the NRC staff for use in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in χ/Q analysis methodology should be reviewed by the NRC staff.

6. ASSUMPTIONS FOR EVALUATING THE RADIATION DOSES FOR EQUIPMENT QUALIFICATION

The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective.

The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used in the evaluation of submittals related to the use of ASTs in radiological consequence analyses at operating power reactors.

[19] The ARCON96 computer code contains processing options that may yield χ/Q values that are not sufficiently conservative for use in accident consequence assessments or may be incompatible with release point and ventilation intake configurations at particular sites. The applicability of these options and associated input parameters should be evaluated on a case-by-case basis. The assumptions made in the examples in the ARCON96 documentation are illustrative only and do not imply NRC staff acceptance of the methods or data used in the example.

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{See the inside front cover of this guide for information on obtaining NRC documents.}

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29. T.J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG-2858, USNRC, November 1982.

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ITEM A-40: SEISMIC DESIGN CRITERIA (REV. 1)

DESCRIPTION

Structures, systems, and components important to the safety of nuclear power plants are required to withstand the effects of natural phenomena such as earthquakes. Broad requirements for earthquake resistance are specified in 10 CFR Parts 50 and 100 and detailed guidance on acceptable ways of meeting these requirements are documented in various regulatory guides. Safety analysis reports for each plant are reviewed in accordance with the review and acceptance criteria described in SRP[11] Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3.

Over the years, there has been an evolution of seismic design requirements and technology. Early nuclear power plants were designed without specific seismic design requirements. In the early 1970s, the requirement for resistance to seismic events was included in the regulations. The state of knowledge has advanced rapidly and the methods of seismic design vary with the vintage of the nuclear power plant. Also, the complex process of seismic design and analysis involved many engineering disciplines: seismic, geotechnical, structural, mechanical, electrical, and nuclear. Each discipline in the design process controlled the design parameters in its domain. As the total seismic design process evolved, two questions emerged: (a) How adequate are the plants in earlier generations with respect to current safety requirements? and (b) What is the margin of safety in the overall seismic design process? This issue was originally identified in NUREG-0371[2] and was later determined to be a USI in NUREG-0510.[186]

CONCLUSION

The objectives of this issue were to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the SRP[11] criteria if changes were found to be justified. Studies under USI A-40 included the following: (1) quantification of conservatism in seismic design; (2) elasto-plastic seismic analysis methods; (3) site-specific response spectra; (4) nonlinear structural dynamic analysis procedures; and (5) soil-structure interaction (SSI).

One key area in seismic design is SSI analysis which is complex and has been controversial in the past. To examine some of the areas of complexity and to obtain an expert consensus, NRC sponsored an SSI workshop in June 1986. The technical areas covered during this workshop were: (1) definition of free field ground motion; (2) ground motion input needed for site-specific SSI analysis; (3) SSI methodology; and (4) experience and experimental verification. Workshop discussions were focused on improving SRP[11] criteria. Reasonable consensus was achieved in the four technical areas and was incorporated into the proposed revisions to SRP[11] Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3.

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Significant results have become available from the joint EPRI/NRC/Taiwan Power Company (TPC) SSI Lotung experiment in Taiwan. These results were presented in an EPRI/NRC/TPC-sponsored workshop in December 1987. The staff therefore formulated specific questions on the Lotung results and solicited comments on them during the public comment period. The resolution of public comments (NUREG/CR-5347)[1238] helped the staff to finalize a position on SSI which was reflected in the revised SRP[11] Section 3.7.2, published as part of the final resolution of USI A-40.

Although some older sites were designed to seismic criteria less rigorous than current requirements, significant upgrading has been or will be achieved by the Systematic Evaluation Program conducted on the oldest plants, the implementation of USI A-46, and by staff bulletins and information notices such as IE Bulletin 79-02,[1239] IE Bulletin 79-14,[1240] and IE Bulletin 80-11.[1241] The staff has therefore concluded that backfit of the proposed seismic design provisions is not necessary except for the design of safety-related large, above-ground tanks at some plants.

The implementation of USI A-46 will result in the review of large, above-ground tanks at about 70 of the older plants. The remainder of the plants fall into two groups: (1) plants that were subject to licensing review by the staff after about 1984; and (2) plants that were reviewed by the staff during the period beginning in the latter part of the 1970s up to 1984. For the plants in the first group, the NRC staff licensing review confirmed that no further action was needed. A survey of the plants in the second group was conducted by the NRC and it was found that tanks for many of these plants were designed using the new criteria. However, the staff was unable to determine the status of large tanks at four sites (Watts Bar, Callaway, Wolf Creek, and Harris) and information request letters were issued to these licensees.

Early activities on USI A-40 consisted of specific technical studies which concentrated on improvements in seismic design criteria. A technical overview and specific recommendations for changes to seismic design criteria were documented in NUREG/CR-1161.[1242] The value/impact assessment for the proposed changes was documented in NUREG/CR-3480.[1243] Based on the recommendations made in NUREG/CR-1161, NUREG/CR-3480, additional staff work discussed in the regulatory analysis (NUREG-1233),[1244] and resolution of public comments (NUREG/CR-5347), the staff revised SRP[11] Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3. These SRP sections are for use in the review of future CPs, PDAs, FDAs, and combined CP/OL applications under 10 CFR 52. In addition to the SRP revisions, the staff will review the seismic adequacy of the large, above-ground vertical tanks at the four nuclear stations outlined above. A discussion of the basis for the selection of these sites is included in NUREG-1233. If the licensee reponses to the NRC's request indicate that these tanks do not meet the proposed criteria, plant-specific backfits will be considered under 10 CFR 50.109. The Commission was informed of the staff's resolution in SECY-89-296.[1245] Thus, the issue was RESOLVED and new requirements were established.[1246]

State of Vermont

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Department of Environmental Conservation

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Department of Forests, Parks and Recreation
Department of Environmental Conservation
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August 26, 2004

William Sherman
State Nuclear Engineer
Department of Public Service
Montpelier, Vermont

Re: Probability of Earthquake Induced Ground Accelerations at Vermont Yankee

Dear Mr. Sherman:

As per your request, the following is information for seismic considerations at Vermont Yankee (VY).

VY received a license to operate in 1972 and is designed for operation to conform to Atomic Energy Commission requirements of April 1968. The maximum ground accelerations at the site are specified as a 0.14g (14% of the acceleration of gravity) Safe Shutdown Earthquake (SSE) and a 0.07g (7% of the acceleration of gravity) Operating Basis Earthquake (OBE). In the late 1990's, the core shroud repair design utilized a USNRC Regulatory Guide 1.60 rev.1 (1973) response spectrum input for the repair seismic analysis.

Since 1972, the science of earthquake prediction has advanced. Probabilistic earthquake analysis that looks at all possible events at once of a certain size in a region are analyzed to predict accelerations at a given locality i.e. Vermont Yankee. A reanalysis of the earthquake catalogue used to make predictions in the northeast has occurred since 1972. Strong motion instruments have been placed in the eastern United States to record response spectra for eastern events (Example - the 6.2 magnitude event of Nov 25, 1988 in Saguenay, Quebec). The following are listing of various earthquake risk analysis approaches that are relevant to predictions in southeastern Vermont.

1972 – VY built using the response spectra from a 1952 Taft, California event.

1980's – USGS develops first probabilistic seismic hazard maps

1995 - "A Report on the Seismic Vulnerability of the State of Vermont", John Ebel et al, Weston Observatory of Boston College – contains Horizontal Peak Ground Acceleration maps

1996 – USGS Probabilistic National Seismic Hazard Maps – Guidance maps of record (minor updates through 2003)

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2 Pages

The USGS interpolated probabilistic 1996 ground motions for a latitude-longitude in the town center of Vernon are as follows:

LOCATION 42 46 35 Lat. -72 31 08 Long.

The interpolated probabilistic ground motion values, in %g:

	10%PE in 50 yr	5%PE in 50 yr	2%PE in 50 yr
PGA	3.96	6.64	12.33
0.2 sec SA	9.24	14.97	26.20
0.3 sec SA	7.57	11.90	20.45
1.0 sec SA	2.78	4.64	8.38

(PGA) - Peak Ground Acceleration, (PE) - Probability of Exceedance, (SA) - Spectral Acceleration, (sec) - Wave Period in Seconds

Since 1999, a Vermont seismic consideration in geotechnical design uses the BOCA National Building Code (1996) in structural designs. The seismic provisions in BOCA include Peak Ground Accelerations (PGA) comparable to that by the USGS for a 500-year return period earthquake event (10% probability of exceedance in 50 years). Though not adopted in Vermont yet, the International Building Code (IBC) is the governing code in 26 states, the District of Columbia and for the Department of Defense. The IBC recommends peak ground accelerations comparable to those by the USGS for a return period of 2500 years (2% probability of exceedance in 50 years).

With NRC "Power Uprate" review at VY, the State is considering issues that are of concern to Vermonters. The following question can be posed - Is there a discrepancy between SSE and OBE accelerations in the license and newer seismic hazard analysis techniques and predictions to warrant a review of critical design elements that relate to seismic safety at VY?

With the above information, the concern would center on short period energy in relation to an IBC building code return period of 2500 years. Short period earthquake induced energy may influence design elements such as shorter structures and piping in nuclear facilities. Predicted shorter period accelerations for 0.2 and 0.3 second period waves in Vernon are 26.20% of the acceleration of gravity (0.2620 g) and 20.45% of gravity (0.2045g). Viewing the numbers of interest, the 1996 shorter period accelerations exceed the 0.14 SSE considerations in the license. This apparent discrepancy warrants further attention and consideration.

Sincerely,

Laurence R. Becker

Laurence R. Becker
Vermont State Geologist

Cc: Elizabeth McLain, Secretary ANR
Jeffery Wennberg, Commissioner DEC
David O'Brien, Commissioner PSD
Albee Lewis, Director of Emergency Management

NUREG-0585

TMI-2 Lessons Learned Task Force Final Report

Manuscript Completed: October 1979
Date Published: October 1979

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington. D. C. 20555

ABSTRACT

In its final report reviewing the Three Mile Island accident, the TMI-2 Lessons Learned Task Force has suggested change in several fundamental aspects of basic safety policy for nuclear power plants. Changes in nuclear power plant design and operations and in the regulatory process are discussed in terms of general goals. The appendix sets forth specific recommendations for reaching these goals.

CONTENTS

CONTENTS (Continued)

TMI-2 LESSONS LEARNED TASK FORCE
FINAL REPORT

1. INTRODUCTION

In May 1979, the Office of Nuclear Reactor Regulation formed an interdisciplinary team of engineers and scientists from various offices of the U.S. Nuclear Regulatory Commission to begin work on the identification and evaluation of safety concerns originating from the accident at Three Mile Island Unit 2 (TMI-2). In July 1979, this team, the TMI-2 Lessons Learned Task Force, issued NUREG-0578 ("TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," Ref. 1) recommending short-term actions to be taken on operating plants and on pending license applications. These short-term recommendations are now being implemented.

In contrast to the short-term recommendations in NUREG-0578, which were of a more narrow, specific, and urgent nature, this report deals with safety questions of a more fundamental policy nature regarding nuclear plant operations and design and the regulatory process. The report addresses these topics in three chapters; each chapter identifies policy elements the Task Force considers to be important and in need of change or improvement. The discussions in these chapters are goal oriented rather than prescriptive in nature, since there may be a number of ways in which the objectives can be achieved. Some objectives would cause significant changes in the nuclear industry and in the regulatory process and should be considered deliberately when choosing the best means of implementation. For others, particularly those related to operations, actions should be initiated without delay since they would introduce a needed and stepwise improvement in safety.

To stimulate discussion and speed the deliberative process, the Task Force has developed a number of specific recommendations toward accomplishing the policy objectives and safety goals described in this report. The specific recommendations are summarized in Appendix A. The Task Force considers the thrust of the modifications it has outlined to be of fundamental importance to nuclear safety, and urges that immediate steps be taken to complete the deliberative process and initiate implementation of these specific recommendations. We envision the deliberative process to include review by the Advisory Committee on Reactor Safeguards; formulation of an action plan by the Office of Nuclear Reactor Regulation in consultation with the Offices of Inspection and Enforcement, Nuclear Regulatory Research, and Standards Development; and approval of the action plan by the Commission. We urge that the action plan address all of the specific recommendations in Appendix A, but we recognize that some may be improved upon in the course of staff, ACRS and Commission review.

We believe that the technical foundation for our specific recommendations is solid, but the recommendations could be affected by the results of studies and investigations that continue inside and outside of the NRC, especially because our scope of responsibility has been narrow in comparison to some of those other efforts. Therefore, the management of NRC will have to exercise some balancing of interests in deciding upon which actions to take now and which actions to study further before regulatory requirements are promulgated. Two

especially important considerations in this balancing of interests, in addition to the improvements in safety inherent in our recommendations, are the need to give prompt and careful consideration to the recommendations of the President's Commission on Three Mile Island and the need to recognize that the bulk of Federal and industry resources are already committed to the timely implementation of shorter term requirements flowing from reviews of the TMI-2 accident and other safety requirements of NRC. The Task Force has given some thought to these factors in developing its suggestions of ways in which implementation could proceed on the specific recommendations in Appendix A. Our judgments on the timing of implementation are stated within the recommendations themselves.

The principal conclusion of the Task Force is that, although the accident at Three Mile Island stemmed from many sources, the most important lessons learned fall in a general area we have chosen to call operational safety. This general area includes the topics of human factors engineering, qualification and training of operations personnel; integration of the human element in the design, operation, and regulation of system safety; and quality assurance of operations. Specifically, the primary deficiency in reactor safety technology identified by the accident was the inadequate attention that had been paid by all levels and all segments of the technology to the human element and its fundamental role in both the prevention of accidents and the response to accidents. Thus, our policy recommendations and our specific ideas for stimulating and accomplishing change concentrate heavily on operations reliability and the associated design and licensing review measures that support or augment operations reliability. But an important qualifier must be added to this conclusion. That is, if the basic responsibility for public safety is to remain in the private sector, in the hands of the individual licensees for commercial nuclear power plants, then significant change in the attention to operations reliability must take place in the licensed industry. Operations is a "hands-on" concept and high operations reliability can only be achieved in practice by those responsible for "hands-on" functions.

The Task Force has given considerable thought to the basic mission of reactor regulation after Three Mile Island. We are not alone in these efforts; many people have called for a clearer articulation of NRC's role and mission since March 28, 1979. However, the Commission and this Task Force recognized soon after the accident that there was a compelling need for short-term, immediate consideration of presently operating plants and steps that needed to be taken to increase their safety. The results of our short-term work and the various other efforts within the NRC and industry have undoubtedly initiated needed improvements in nuclear reactor safety. But much more is needed beyond these reactionary steps. The Task Force acknowledges and appreciates the unique opportunity it has to stand back and look broadly at the past and the future of reactor safety regulation. This opportunity has led us to a critical scrutiny of NRC safety policy. What we have found is that prescriptive and narrow licensing requirements only add to the quiltwork of regulatory practice and do little to directly address the nation's heightened concern for the safety of nuclear power plants. What seems to be missing is the common denominator of an articulate and widely noticed national nuclear safety policy with which to bind together the narrow and highly technical licensing requirements. The Commission has alluded to a more definitive safety policy by taking actions that in effect say, "no more Three Mile Islands." But the feasibility and the adequacy of such a policy must be critically examined and an opportunity

should be provided for thorough and widespread public input. Such dialogue and debate at a widely comprehensible level will enable the NRC to realize its leadership role in nuclear safety and diminish our partially deserved image as a reactionary body that is both defensive and apologetic of nuclear power. The need to articulate our basic safety policy is compelling. It need not wait for a new statutory mandate, and it should not be a de facto stepchild of future events.

2.3.3 Training

In determining the qualifications of personnel, academic education, experience, health, and training are taken into account. A principal element in achieving the desired level of competence is training. Once a level of competence is achieved, it must be continually reinforced. Thus, training should be an ongoing process. Utility management must assure itself that personnel occupying all positions are able to perform the tasks required of them in normal and accident situations. The Task Force recommends that each licensee should be required to review its training program, using a position task analysis for all operations personnel, and to justify the acceptability of training programs on the basis that they provide sufficient assurance that safety-related tasks will be carried out effectively (Recommendation 1.2). It is expected that completion of this review will lead to the identification and correction of weaknesses where they exist in present training programs. We also see, in both government and industry, that there is a need to include the expertise of professional educators in improving reactor operations training programs (see Recommendation 1.5).

2.3.4 Emergency Operating Procedures

The use of properly prepared procedures in plant operations is another important ingredient in the matrix of operational safety. Attention must be given to both normal and emergency operating procedures. Although the Task Force recognizes the importance of normal operating procedures, it has, because of limits on time and expertise, directed its attention primarily to emergency operating procedures. Emergency operating procedures should consider system interactions and be written in such a manner that they are unambiguous and useful in crisis control. They should be based on thorough engineering evaluation and realistic analyses of the dynamic response of the nuclear power plant. The Task Force has found the NRC review process for emergency procedures to be inadequate and is recommending that present practice be changed to provide for interdisciplinary review of emergency procedures as part of the operating license review process (see Recommendation 4). Past practice was not sufficient because it did not specifically investigate the compatibility of emergency procedures with the design bases of the systems involved, nor was the discipline of human factors involved. The reviews should also include consideration of experience outside the commercial nuclear industry in the use of written procedures for crisis mitigation.

2.3.5 Working Environment and Operational Aids

The first line of reliance for safe operation of a nuclear power plant is the reactor operators and their immediate control room supervisor. Operator action in accordance with improved training and better operating procedures can prevent a number of challenges to safety systems and thus prevent potential accidents. In the event that safety systems fail and procedures do not apply, the operators are also the last line of reliance; i.e., they are the key component in contingency decisions and accident mitigation strategies if the design basis for the plant is exceeded. To diagnose and respond to plant disturbances, the operators must be well-qualified and their human actions must be integrated with the machine actions of the plant design. Control systems and related displays should also be integrated and easily identified

at all times during operation of the facility." For single-unit power stations, the staff requires the shift crew to include at least one licensed senior reactor operator, two licensed reactor operators, and two additional operators (auxiliary operators) during reactor operation. For multiple-unit power stations with separate control rooms, the staff also requires the shift crew to include at least one licensed senior reactor operator and two licensed reactor operators for each operating reactor. For multiple-unit power stations with a common control room, the staff permits a reduction of licensed reactor operators to one per unit plus one additional reactor operator with the other requirements remaining the same. However, the staff does not require the presence in the control room at all times of two licensed operators and the senior reactor operator. In developing the revision to the regulations, consideration should be given to requiring the presence in the control room at all times during normal operations of two reactor operators and one senior reactor operator. Provisions for tours of the plant by operators will probably need to be made if this staffing proposal is adopted.

3. WORKING HOURS

Each licensee should be required to review and revise within 90 days the plant administrative procedures to assure that a sound policy is established covering working hours for reactor operators and senior reactor operators. It is recognized that this is a complex subject involving other interests (e.g., labor unions). The NRC staff should assure that the subject is addressed in a comprehensive manner by all licensees and that the other interests not be allowed to interfere with the basic safety interest. As general guidance, it is expected that licensees' administrative procedures will make it unlikely that personnel would have to be used for more than two consecutive work periods in excess of 12 hours and that a 12-hour rest period would be required between work periods. In the event that special circumstances arise that would cause extended periods of work in excess of 12 hours for more than two consecutive days, such work should be authorized by the Station Manager with appropriate documentation of the cause. Indications aside from Three Mile Island lead the Task Force to conclude that this step must be taken to reasonably assure that individuals are in proper physical condition to perform work at nuclear power plants.

4. EMERGENCY PROCEDURES

Emergency operating procedures for all nuclear power plants should be reviewed by the NRC. The review should be conducted by interdisciplinary review groups comprising I&E inspectors and NRR technical reviewers knowledgeable in system design, accident analysis, operator training, theories of education and crisis management, human factors, and the underlying technical bases for licensing. Special attention should be paid to the recent advice of the ACRS on the style and content of emergency procedures. A safety evaluation regarding the adequacy of the emergency procedures should be issued at the conclusion of the review. Previous NRR reviews and I&E reviews of emergency operating procedures did not specifically investigate their compatibility with the design bases of the systems involved nor was the discipline of human factors included.

This action will require a considerable expenditure of resources and its phasing needs to be carefully considered. It may be satisfactory to limit the general application of this recommendation to new operating licenses for the next year or so. These initial few reviews by the staff, with oversight by the ACRS, will provide the time and experience necessary for the staff and industry to develop and agree upon acceptance criteria for the development, formatting, and future review of all emergency operating procedures. Upon completion of these acceptance criteria, say within the next two years, a systematic effort by all licensees to review their emergency procedures and revise them as necessary could be conducted more productively than it could today.

5. VERIFICATION OF CORRECT PERFORMANCE OF OPERATING ACTIVITIES

A more effective system of verifying the correct performance of operating activities is needed to provide a means of reducing human errors and improving the quality of normal operations, thereby reducing the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system should include automatic system status monitoring and human verification of operations and maintenance activities independent of the people performing the activity.

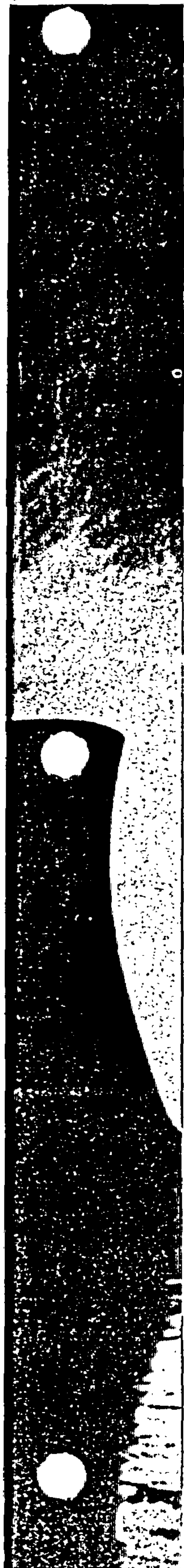
The Task Force recommends that automatic status monitoring be required by a decision to backfit Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," to plants not already required to meet it. Furthermore, the design to satisfy the objectives of the guide should be flexible and capable of accepting additional monitoring functions at a later date.

The implementation of Regulatory Guide 1.47, although reducing the extent of human verification of operations and maintenance activities, does not eliminate the need for such verification in all instances. Therefore, each licensee should be required to review his procedures for maintenance, test, surveillance and other normal plant operations activities (1) to delineate each activity that requires independent verification because of its importance to safety, (2) to identify the personnel responsible for conducting the verification, and (3) to describe the method of documenting performance of the verification process. The results of this work should be submitted to NRC within six months for use in the development of minimum acceptance criteria for operations verification procedures, probably in the form of a Regulatory Guide. The procedures adopted by the licensees should contain two phases; namely, before and after installation of status monitoring equipment in conformance with Regulatory Guide 1.47.

6. EVALUATION OF OPERATING EXPERIENCE

6.1 <u>Nationwide Network</u>

An integrated NRC-utility program to evaluate operating experience should be established. Action within the NRC has been initiated to establish an Office of Operational Data Analysis and Evaluation to provide agency-wide coordination and an overview of all operational data analysis-related activities performed within the line offices of NRC. The nuclear industry, through NSAC and INPO,



three mile island

VOLUME I

A REPORT TO THE COMMISSIONERS AND TO THE PUBLIC

mitchell rogovin
director

george t. frampton, jr.
deputy director

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NUCLEAR REGULATORY COMMISSION
SPECIAL INQUIRY GROUP

contents

5 GREATER APPLICATION OF HUMAN FACTORS ENGINEERING, INCLUDING BETTER INSTRUMENTATION DISPLAY AND IMPROVED CONTROL ROOM DESIGN

In 1975 one of the NRC staff's leading safety experts, Stephen Hanauer, in a memorandum to Commissioner Victor Gilinsky setting forth the major problems in safety that should be addressed by the new NRC, said: "Present designs do not make adequate provision for limitations of people." The President's Commission found that both the industry and the NRC have "failed to recognize sufficiently . . . that the human beings who manage and operate the plants constitute an important safety system." A senior B&W official put it a different way: he told us that the industry had done a fine job in engineering safety equipment, but that good engineering also meant designing for people—that the industry had fallen down in "bringing operators within the design envelope."

During the period in which most large nuclear plants have been designed, the nuclear industry has paid remarkably little attention to one of the best tools available for integrating the reactor operator into the system: the relatively new discipline of "human factors." Human factors engineering was born of military needs during World War II and has since blossomed in the aerospace, defense and aircraft industries. But nuclear utilities, vendors and architect-engineer firms have done very little to incorporate such learning into their designs, and the NRC has done virtually nothing to require them to do so.

This failure reflects the preoccupation of the industry and the regulatory agency with hardware systems. The NRC gives short shrift in the design safety review process to determining how well operators will be able to diagnose abnormal events, based on what they see on their instruments, and respond to them. In part, the failure is also due to a lack of expertise. The agency has no office or staff members charged with examining the interaction between operators and design systems. Before the Three Mile Island accident, virtually no one in the agency was knowledgeable about such matters, and the agency was making no effort to seek out such people.

The pernicious effects of this attitude on the safety review of design are clear. Our investigation found that the TMI-2 plant was substantially more dependent than other designs upon operator action to prevent a routine loss-of-feedwater transient from turning into a possible accident under circumstances such as a stuck-open valve. The specific "operator sensitivities" of plants similar to TMI-2 revealed by the accident have since been ameliorated by changes in control logic and by the setting of different "setpoints" for automatic operation of certain equipment. The point is that this sensitivity—and undoubtedly the sensitivity of other designs to operator action in other types of "transients"—has never received much attention from the industry or the NRC.

VY CALCULATION CHANGE NOTICE (CCN)

CCN Number: 4 Calculation Number: VYC-0808 Rev. No. 6

Calculation Title: Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident or Anticipated Transients Without Scram

Initiating Document: VYDC 2003-008
VYDC/MM/TM/Spec. No./ other

Safety Evaluation Number: N/A

Superseded Calculation: N/A Superceded by: N/A

Implementation Required: ☒ Yes ☐ No

Computer Codes: N/A

Reason for Change: Residual Heat Removal (RHR) and Core Spray (CS) Pump NPSH must be evaluated for the affect of increased suppression pool temperatures, resulting from the Extended Power Uprate (EPU).
Description of Change: NPSH evaluation of the RHR and CS pumps is performed for the following events for which EPU suppression pool temperature/pressure data is presently available. • DBA-Loss of Coolant Accident (LOCA), short term and long term • Anticipated Transients Without Scram (ATWS)
Technical Justification for Change: The reactor core decay heat increases as core power in increased. This heat is transferred to the suppression pool during accidents, transients and other events, which result in peak suppression pool temperatures that are higher than at the existing (pre-EPU) power level. Since increased pool temperature reduces available NPSH, the adequacy of available NPSH must be demonstrated for each event that credits pumping suppression pool water at elevated temperatures. If available NPSH is inadequate, then torus overpressure must be credited.
Conclusions: There is adequate NPSH available for operating the RHR and CS pumps at EPU conditions for the DBA-Loss of Coolant Accident (LOCA), short term, without crediting torus overpressure. Torus overpressure must be credited for operating the RHR and CS pumps at EPU conditions for the following events in order to achieve adequate NPSH available: • DBA-Loss of Coolant Accident (LOCA), long term • Anticipated Transients Without Scram (ATWS) The overpressure credit required for LOCA bounds that required for ATWS. The results of this CCN will provide input to the PUSAR (Ref: 12) for the RHR and CS NPSH evaluation and will alter input to calculation VYC-1628 (Ref: 13) to address the need for crediting torus overpressure in the calculation of NPSH available. Note that calculation VYC-1628 may be superseded by GE EPU Analysis. The need for crediting torus overpressure in the RHR and CS NPSH evaluation, shall also be addressed in the SADBD (Ref: 14), UFSAR (Ref: 15), and system DBDs RHR (Ref: 16) and CS (Ref: 17). Note that use of overpressure credit must be approved by the NRC as part of EPU. No specific 50.59 Screening/Evaluation is required for this CCN since all EPU design changes and associated 50.59 documentation will be part of VYDC-2003-008. See Section 5.0 for Summary of Results

Note: VYAPF 0017.07 should be included immediately following this form.

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Are there any open items in this CCN? ☐ Yes ☒ No

Prepared By/Date	Interdiscipline Review By/Date	Independent Review By/Date	Approved By/Date
Frank Capuano /SW/ 08-08-03 F. Capuano	N/A	Edward Lind / SW / 08-12-03 Edward Lind	for OE Yori 8/14/03

Final Turnover to DCC (Section 2): Accepted by BC Shifer 8/14/03

1) All open items, if any, have been closed.

2) Implementation Confirmation (Section 2.3.4)

☐ Calculation accurately reflects existing plant configuration, (confirmation method indicated below)

☐ Walkdown ☐ As-Build input review ☐ Discussion with ______________________ (print name)

OR

☐ N/A, calculation does not reflect existing plant configuration

3) Resolution of documents identified in the Design Output Documents Section of VYAPF 0017.07 has been initiated as required (Section 2.3.6, 2.3.7)

______________ / ______________ / ______________
Print Name Signature Date

Total number of pages in package including all attachments 53

VYC-0808 Revision 6
CCN #4

Note: VYAPF 0017.07 should be included immediately following this form.

Place this form in the calculation package immediately following the Title page or CCN form.

VYC-0808 / 04	6	N/A	N/A
VY Calculation/CCN Number	Revision Number	Vendor Calculation Number	Revision Number

Vendor Name: N/A PO Number: N/A

Originating Department: Fluid Systems

Critical References Impacted: ☒ UFSAR ☒ DBD ☐ Reload. "Check" the appropriate box if any critical document is identified in the tables below.

EMPAC Asset/Equipment ID Number(s): P-10-1A/B/C/D, P-46-1A/B

EMPAC Asset/System ID Number(s): 10, 14

Keywords: Residual Heat Removal (RHR), Core Spray (CS), Net Positive Suction Head (NPSH), Loss of Coolant Accident (LOCA), Extended Power Uprate (EPU) Anticipated Transients Without Scram (ATWS)

For Revision/CCN only: Are deletions to General References, Design Input Documents or Design Output Documents required? ☐ Yes† ☒ No

Design Input Documents and General References - The following documents provide design input or supporting information to this calculation. (Refer to Appendix A, sections 3.2.7 and section 4)

* Reference #	** DOC #	REV #	***Document Title (including Date, if applicable)	Significant Difference Review ††	**** Affected Program	Critical Reference (✓)
1	GE-VYNPS-AEP-177	N/A	VYNPS EPU T0400: Suppression Pool Temperature/Pressure Response to DBA-LOCA (4/5/030)			
2	VYC-1924	0	DE&S Calc. DC-A34600-006. Vermont Yankee ECCS Suction Strainer Head Loss Performance Assessment, RHR and CS Debris Head Loss Calculations			
3	VYC-0808 / CCN 03	6	Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident			
4	VYC-0808	6	Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident			
5	GE-NE-0000-0016-3831-01	0	YYNPS EPU Task T0902: Anticipated Transients Without Scram			
6			NOT USED			
7			ASME Steam Tables, 1967 IFC Formulation for Industrial Use			
8	ERC No. 2003-027	N/A	Debris Source Terms Appropriate for Power Uprate Evaluations of ECCS NPSH (5/13/03)			
9	RHR	1/IC16	Design Basis Document for Residual Heat Removal System			✓
10	VYC-1924 / CCN 02	0	DE&S Calc. DC-A34600-006. Vermont Yankee ECCS Suction Strainer Head Loss Performance Assessment, RHR and CS Debris Head Loss Calculations			
11	VYC-1524	3	Containment and RPV Volume Calculations			

Design Output Documents - This calculation provides output to the following documents. (Refer to Appendix A, section 5)

* Reference #	** DOC #	REV #	Document Title (including Date, if applicable)	**** Affected Program	†††Critical Reference (✓)
12	NEDC-33090P	0	Safety Analysis Report For VYNPS Constant Pressure Power Uprate (PUSAR)		
13	VYC-1628	0	Torus Temperature and Pressure Response to Large Break LOCA and MSLB Accident Scenarios (may be superseded be GE EPU analysis)		
14	SADBD	2/IC2	Topical Design Basis Document for Safety Analysis		✓
15	UFSAR	17	Updated Final Safety Analysis Report		✓
16	RHR	1/IC16	Design Basis Document for Residual Heat Removal System		✓
17	CS	0/IC10	Design Basis Document for Core Spray System		✓

* Reference # - Assigned by preparer to identify the reference in the body of the calculation.

** Doc # - Identifying number on the document, if any (e.g., 5920-0264, G191172, VYC-1286)

*** Document Title - List the specific documentation in this column. "See attached list" is not acceptable. Design Input/Output Documents should identify the specific design input document used in the calculation or the specific document affected by the calculation and not simply reference the document (e.g., VYDC, MM) that the calculation was written to support. If a DBD is used as a general reference, include the most current interim change number after the title.

**** Affected Program - List the affected program or the program that reference is related to or part of.

† If "yes," attach a copy of "VY Calculation Data" marked-up to reflect deletion (See Section 3.1.8 for Revision and 5.2.3.18 for CCNs).

†† If the listed input is a calculation listed in the calculation database that is not a calculation of record (see definition), place a check mark in this space to indicate completion of the required significant difference review. (see Appendix A, section 4.1.4.4.3). Otherwise, enter "N/A."

††† If the reference is UFSAR, DBD or Reload (IASD or OPL), check Critical Reference column and check UFSAR, DBD or Reload, as appropriate, on this form (above).

Table of Contents

Table of Contents (cont'd)

** NOTE: All VY comments on this CCN have been resolved and incorporated as applicable. The resolutions have been accepted by VY.*

1.0 Objective

The objective of this CCN is to evaluate the adequacy of the available NPSH for the Residual Heat Removal (RHR) and Core Spray (CS) pumps to operate at Extended Power Uprate (EPU) conditions. This includes identification of the amount of suppression pool (torus) overpressure required, if any, to maintain adequate NPSH.

The reactor core decay heat increases as core power in increased. This heat is transferred to the suppression pool during accidents, transients and other events, which result in peak suppression pool temperatures that are higher than at the existing (pre-EPU) power level. Since increased pool temperature reduces available NPSH, the adequacy of available NPSH must be demonstrated for each event that credits pumping suppression pool water at elevated temperatures.

EPU data is presently available for the following events, and as such the EPU impact of these events on NPSH is included in this CCN.

- DBA-Loss of Coolant Accident (LOCA), short term and long term
- Anticipated Transients Without Scram (ATWS) which include:
  - Main Steam Isolation Valve Closure (MSIVC)
  - Pressure Regulator Fail Open – Maximum Steam Demand (PRFO)

Additionally, a basis for readily determining overpressure requirements, when performing RHR and CS pump NPSH evaluation for any other events which cause elevated suppression pool temperatures, will be provided. This will be in the form of a family of curves profiling overpressure required vs. pool temperature.

2.0 Methodology

General

The methodology for determining the NPSH available (NPSHa) for a given event and temperature is the same as that developed in VYC-0808 Rev 6 and presented in Table 1 of that calculation (Ref: 4). The NPSH required (NPSHr) is also per VYC-0808 Rev 6 and is discussed in detail in Section 4.0 of this CCN.

Note that prior revisions and CCNs of this calculation have considered torus pressure to remain at atmospheric pressure in the evaluation of NPSHa. However, because of the increased pool temperature and resulting increased vapor pressure, there will not be adequate NPSHa for some events without taking credit for some torus air space pressure. If NPSHa is inadequate, then the necessary torus air space pressure, above atmospheric pressure, (overpressure) will be calculated to yield adequate NPSHa.

For a high temperature, time dependent event, such as a LOCA, the NPSH will be evaluated over the time-temperature profile of the event in lieu of just at the maximum temperature. This will allow development of a time dependent profile for required overpressure.

Pump Suction Strainer Head Loss during a LOCA

As documented in ERC No. 2003-027 (Ref: 8), EPU does not affect the debris source terms developed in VYC-1677. Therefore the limiting head loss due to debris loading on the RHR and CS suction strainers remains the same as addressed in VYC-0808 Rev 6 including CCN #3 (Ref: 3).

CCN #3 dispositions the up-to-date suction strainer head loss calculated in CCN #2 of VYC-1924 Rev 0 (Ref: 10) which in turn is based on the up-to-date debris source term information per VYC-1677 Rev 0 CCN #3. Note that the debris loading from calc VYC-0808 Rev 6, as extracted from calc VYC-1924 Rev 0 (Ref: 2), is slightly larger than that documented in CCN #3. For conservatism, the larger debris loading is used.

It should be noted that the limiting head loss due to debris loading on the RHR and CS suction strainers is calculated at a specific temperature. Strainer head loss is essentially inversely proportional to fluid temperature as documented in the sensitivity evaluation in VYC-1924 Rev 0 (Ref: 2). Therefore the calculated limiting head loss, described above, will be conservatively used for all fluid temperatures greater than or equal to that used in the calculation of the limiting head loss. For lower temperatures, the head loss will be increased in proportion to the decrease in temperature.

General Profile – Overpressure Required vs. Pool Temperature

In addition to specifically addressing the LOCA and ATWS events, a general profile of Overpressure Required vs. Pool Temperature, for both the RHR and CS pumps, is provided. This profile is intended to serve as the basis for determining overpressure requirements when performing RHR and CS pump NPSH evaluation for any other events, which cause elevated suppression pool temperatures, without strainer debris loading.

This profile is provided for a range of pump flows consistent with the maximum flows evaluated for LOCA and the flows for which NPSH data is provided by the pump vendor.

Only the clean strainer loss is accounted for in the profile, as debris loading on the pump suction strainer is not postulated.

The suppression pool volume is conservatively taken to be at its minimum volume (68,000 ft^3), as addressed in Section 2.4 of Calc VYC-0808 (Ref: 4). This conservatively minimizes the suction elevation head.

There is no additional NPSH margin included in this evaluation.

3.0 Assumptions

1. The RHR flow rate evaluated for ATWS is not specified in GE Task Report T0902 (Ref: 5). Therefore, based on the system design basis provided in Section 2.3.2 of the RHR DBD (Ref: 9) the analyzed flow rates are conservatively taken as 7400 gpm for one RHR pump and 14,200 gpm for two RHR pumps (DBD Section 2.3.2.1.2). These are the same flows rates used for the LOCA evaluation and are greater than the 6400 gpm (minimum) specified in DBD Section 2.3.2 for suppression pool cooling. Suctions losses and strainer losses are conservatively larger with larger flow rates.

2. The torus is considered to be at atmospheric pressure for the purpose of calculating the specific volume of the pool water in the evaluation performed in Section 4.4, General Profile – Overpressure Required vs Pool Temperature. The impact of small variations in pressure is insignificant. An example follows:
   v @ 180°F/14.7 psia = .016510 ft^3/lb
   v @ 180°F/20.7 psia = .016509 ft^3/lb

4.0 Analysis

As stated in Section 2.0, the methodology for determining the NPSHa is the same as that developed in VYC-0808 Rev 6 and presented in Table 1 of that calculation (Ref: 4). The following terms are used in the evaluation.

NPSHa (ft) = net positive suction head available without overpressure credit
(14.7-Pg)(144vf)+Z-hf-hs-hd

where:

Z (ft) = suction elevation head

hf (ft) = suction line losses

hs (ft) = clean strainer losses

hd (ft) = strainer debris losses

Pg (psia) = vapor pressure @ torus temperature

vf (ft^3/lb) = specific volume @ torus temperature and pressure

Pg and vf are obtained from ASME Steam Tables 1967 Formulation (Ref: 7)

NPSHr (ft) = net positive suction head required.

It should be noted that the NPSH required data provided by the pump vendor, as documented in Figures 2.1-1 and 2.2-1 of Attachment 3 to calculation VYC-0808, is actually *Allowable Operating Periods @ NPSHa Specified.* Allowable hours of operation at specified NPSHa values are identified for a range of flows. For this CCN, the NPSHa specified in these Figures is taken as the NPSHr at a given operating time.

Q (gpm) = pump flow rate

OPR (psig) = Overpressure Required
(NPSHr – NPSHa)/(144*Vf)

For those profile points where there is inadequate NPSH, when considering the suppression pool pressure to be atmospheric (14.7 psia), OPR is the amount of suppression pool pressure required to make NPSHa (ft) equal to NPSHr (ft).

OPA (psig)– Overpressure Available
The suppression pool pressure available, above atmospheric, for a given event and time.

OPC (psig)– Overpressure Credit Taken
The overpressure credited in the evaluation of NPSH. Engineering judgement is used to select the credit to be greater than the OPR, by a reasonable amount, and less than the OPA.

Detailed discussion of the above terms, as applicable for each event, is provided in the subsections that follow.

4.1 LOCA – Short Term

The temperature and pressure (T/P) profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-177 (Ref: 1). The short term data is provided from 0-600 seconds.

The evaluation of NPSH is documented in Table 4.1 using the peak pool temperature of 165.1°F which occurs at 600 seconds with a corresponding pool pressure of 17.75 psia. The peak temperature results in the largest vapor pressure and lowest NPSHa. Note that the temperature at lowest pool pressure is 162.5°F / 17.56psia. At this temperature the gain in vapor pressure more than offsets the reduction in pool pressure, therefore the 165.1°F case governs. The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for CS and RHR. Further discussion of selected terms is presented below.

<u>Suction Elevation Head, Z</u>
The values of Z for RHR and CS (12.30' and 12.47'respectively) as calculated in Section 3.5 of VYC-0808 are conservatively used in this evaluation. The suction elevation head is based on the water elevation in the torus. The EPU suppression pool water volume is slightly larger than the existing value used in VYC-0808, which would result in a slight increase in water elevation, and therefore Z.

A water volume comparison at maximum pool temperature is provided below:

	Pre-EPU	EPU
Ref:	(VYC-0808 Rev 6 Section 3.5)	(GE-VYNPS-AEP-177)
Short Term	76,800 cuft	79,370 cuft

<u>Maximum Debris Losses (hd)</u>

1 RHR: CCN #3 (Ref: 3) calculated the limiting head loss as 0.24 ft at 181.7°F and 7400 gpm. Note that this is a slight reduction from the head loss (0.33 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.33 ft at 173°F is used. (Case 1 of Tables 2 and 8 of Ref: 2).

2 RHR: The head loss is taken as .48 ft (Ref: 4) at 170°F (Case 2b of Tables 2 and 8 of Ref: 2) and 14200 gpm.

CS The head loss is conservatively taken as .32 ft (Ref: 4) at 173°F (Case 3d of Tables 2 and 8 of Ref: 2) and 4600 gpm.

Refer to Section 2.0 for application of head loss at temperatures other than those used in its calculation.

NPRHr - CS

Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 4600 gpm an allowable NPSH of 28.0 ft is acceptable between 0 and 7 hrs of operation.

NPRHr - 1 RHR

Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 7400 gpm an allowable NPSH of 23.8 ft is acceptable between 0 and 7 hrs of operation.

NPRHr - 2 RHR

With two RHR pumps operating at a total flow of 14,200 gpm this yields a flow of 7100 gpm per pump.
Also per Figure 2.1-1, the plot shows that at between 0 and 7 hrs of operation, an allowable NPSH of 23.5 ft is acceptable at 7000 gpm and 24.0 ft is acceptable at 7600 gpm.

Interpolating between plotted NPSH values of 23.5 ft @ 7000 gpm and 24.0 ft @ 7600 gpm yields 23.6 ft @ 7100 gpm.

The interpolation equation is developed as documented Section 2.2.2 of VYC-0808 Rev 6 and is 23.0+(Q-6400)/1200

4.1.1 Evaluation

As can been seen from Table 4.1, there is adequate NPSHa and overpressure is not required.

4.2 LOCA – Long Term

The temperature and pressure (T/P) profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-177 (Ref: 1). The long term data is provided from 0-172,800 seconds.

The evaluation of NPSH is documented in Table 4.2 using a selected T/P points representing the long term profile of the suppression pool. The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for the T/P profile of CS and RHR. The evaluated long term flow rates of 7400 gpm (RHR) and 3500 gpm (CS) are consistent with calculation VYC-0808 Rev 6 (Ref: 4). Further discussion of selected terms is presented below.

Suction Elevation Head, Z

The values of Z for RHR and CS (12.40' and 12.57' respectively) as calculated in Section 3.5 of VYC-0808 are conservatively used in the evaluation. The suction elevation head is based on the water elevation in the torus. The EPU suppression pool water volume is slightly larger than the existing value used in VYC-0808, which would result in a slight increase in water elevation, and therefore Z.

A water volume comparison at maximum pool temperature is provided below:

	Pre-EPU	EPU
Ref:	(VYC-0808 Rev 6 Section 3.5)	(GE-VYNPS-AEP-177)
Long term	77,640 cuft	79,470 cuft

<u>Maximum Debris Losses (hd)</u>
1 RHR: Refer to Section 4.1

CS Note that CCN #3 (Ref: 3) documents the up-to-date limiting head loss as 0.19 ft at 181.7°F and 3500 gpm. This is a slight reduction from the head loss (0.21 ft) addressed in Section 3.2 of VYC-0808 Rev 6 (Ref: 4). For conservatism, 0.21 ft at 173°F is used. This is based on a conservative CS flow rate of 4000 gpm. (Case 3b of Tables 2 and 8 of Ref: 2).

<u>NPRHr - CS</u>
Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 3500 gpm the allowable NPSH increases between 7 and 20 hrs of operation and a value of 29.6 ft is acceptable beyond 20 hrs of operation. This maximum value is conservatively used for the entire long term period (>600 sec).

<u>NPRHr - RHR</u>
Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 7400 gpm the allowable NPSH increases between 7 and 100 hrs of operation and a value of 31.7 ft is acceptable beyond 100 hrs of operation. This maximum value is conservatively used for the entire long term period (>600 sec).

4.2.1 Evaluation
As can been seen from Figure 4.2 the overpressure required for RHR envelopes that required for CS and the overpressure varies continuously over time. In order to facilitate reporting and presentation of the overpressure required, an enveloping, stepped, overpressure credit is overlaid on Figure 4.2. Refer to Section 4.0 for discussion on selection of overpressure credit.

Though the long term flow rates are postulated at time 600 seconds (e.g. CS throttled down from 4600gpm to 3500gpm), it is not the intent of this calculation to imply at what time throttling should commence or how much throttling is required. This is a function of the time dependent NPSHr and pool temperature. This calculation conservatively evaluates the maximum NPSHr as occurring over the entire operating period (>600 sec). The actual NPSHr is lower between 0-7 hrs and increases after 7 hrs.

Note that Section 4.4 develops required overpressure for both the CS and RHR pumps as a function of flow, temperature and NPSHr without any debris loading. Refer to Table 4.4 and Figures 4.4-1 to 4.4-4.

4.3 ATWS

Note that NPSH evaluation of the ATWS event was not previously addressed by calculation VYC-0808. The temperature and pressure (T/P) profile for the suppression pool during the ATWS events is developed in GE-Task Report T0902 (Ref: 5).

The evaluated events are MSIVC and PRFO with the peak temperatures and corresponding pressures tabulated by GE in Section 3.3.1.2 of the Task Report. Selected data points are extracted from the included T/P profile plots, Figures 3-10 and 3-12 of the Task Report, and are shown below. The two events have essentially the same temperature pressure profile. For convenience, these are combined into one enveloping event with maximum temperatures and minimum pressure at each time step.

MSIVC			PRFO			Combined		
Time, sec	Temp, °F	Press, psig	Time, sec	Temp, °F	Press, psig	Time, sec	Temp, °F	Press, psig
0	90	0	0	90	0	0	90	0
300	160	6.3	300	157	6.3	300	160	6.3
600	175	8.8	600	168	8.2	600	175	8.2
1000	182	11.2	1000	180	10.7	1000	182	10.7
1300	187	11.5	1300	187	11.8	1300	187	11.5
1724	189	12.3	1838	190	12.5	1838	190	12.3
3000	186	11.9	3000	187	12.1	3000	187	11.9
5000	182	11.2	5000	182	11.3	5000	182	11.2
6000	180	10.8	6000	180	11.0	6000	180	10.8
8000	175	10.0	8000	175	10.2	8000	175	10.0

As documented in T0902, Section 3.2.11 and 3.2.2.2, the suppression pool cooling is based on two loops of RHR operating and an initial pool volume of 68,000 cuft. Note that CS does not operate for ATWS events.

The evaluation of RHR pump NPSH is documented in Table 4.3 for the reduced NPSHr (0-7 hrs) The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for RHR. Further discussion of selected terms is presented below.

Since there are no data points available beyond 8000 seconds. Evaluation of intermediate NPSHr (7hrs – 20hrs) and maximum NPSHr (20hrs – 100hrs) are addressed individually below.

<u>Flow rate – Q (gpm)</u>
The RHR flow rate that was evaluated is not specified in T0902. The flow rates are assumed to be 7400 gpm for one RHR pump and 14,200 gpm for two RHR pumps. Refer to Assumption No. 1 of this CCN.

Evaluation will be performed for both 1 and 2 RHR pump operation.

Suction Elevation Head, Z

The suppression pool volume addressed in the GE report is 68,000 ft^3. As noted in Section 3.5 of Calc VYC-808 the relationship between suppression pool volume and level is documented in Table 4.3-1 of Calc VYC-1254 Rev 3 (Ref: 11). The suppression pool level corresponding to 68,000 ft^3 is 10.88'from VYC-1254.

This water level is 1.05' less than (11.93'-10.88') that used for calculating the RHR short term Z (12.30') in calc VYC-0808. Therefore the adjusted Z for the ATWS evaluation is 12.3'-1.05' = 11.25'.

Maximum Debris Losses (hd)

This term is not applicable for ATWS since there is no high-energy line break (HELB) to dislodge insulation and create debris in the suppression pool.

Minimum NPRHr - 1 RHR (<7hrs)

Refer to Section 4.1

Minimum NPRHr - 2 RHR (<7hrs)

Refer to Section 4.1

Intermediate NPRHr - 1 RHR (20 hrs)

Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. There is an intermediate peak at 20 hrs. This plot shows that at 7400 gpm the allowable NPSH increases between 7 and 20 hrs of operation and a value of 30.5 ft is acceptable at 20 hrs of operation.

Intermediate NPRHr - 2 RHR (20 hrs)

With two RHR pumps operating at a total flow of 14,200 gpm this yields a flow of 7100 gpm per pump.
Also per Figure 2.1-1, the plot shows that the allowable NPSH increases between 7 and 20 hours of operation. At 20 hrs of operation an allowable NPSH of 28.5 ft is acceptable at 7000 gpm and 31.6 ft is acceptable at 7600 gpm.

Interpolating between these NPSH values, at a flow of 7100 gpm, yields 29.0ft @ 7100 gpm

Maximum NPRHr - 1 RHR

Refer to Section 4.2

Maximum NPRHr - 2 RHR

With two RHR pumps operating at a total flow of 14,200 gpm this yields a flow of 7100 gpm per pump.
Also per Figure 2.1-1, the plot shows that an allowable NPSH of 29.5 ft @ 7000 gpm and 32.75 ft @ 7600 gpm is acceptable between 100 and 8000 hrs of operation.

Interpolating between these NPSH values, at a flow of 7100 gpm, yields 30.0 ft @ 7100 gpm.

4.3.1 Evaluation Minimum NPSHr (<7 hrs)

As can been seen from Table 4.3, the results for 2 RHR pump operation provide lower NPSHa than for 1 pump operation and overpressure is required from about 600 to 6000 seconds. An overpressure of 2.4 psig is required to accommodate the peak pool temperature of 190°F at 1828 seconds. A plot of the overpressure required is shown in Figure 4.3. Refer to Section 4.0 for discussion on selection of overpressure credit.

Note that for the same time period, this overpressure credit (2.4 psig) is bounded by that required for LOCA (2.4 to 4.4 psig).

4.3.2 Evaluation of Intermediate and Maximum NPSHr (>7 hrs)

As seen from Figures 3-10 and 3-12 of the Task Report and the data tabulated above, the temperature and pressure profiles decay linearly between 3000 – 8000 seconds at the rate of about 2.4°F and 0.4 psig per 1000 seconds. The available data ends at 8000 seconds (175°F/10.0psig). However the temperature will continue to decay though the rate of decay will decrease as the pool cools.

Included, as part of Table 4.3, is a tabulation of "NPSHa vs. decreasing pool temperature" for 1 and 2 RHR pump operation. Two pump operation is selected for detailed evaluation as it yields an NPSHa about 2.2' less than one pump operation. Since pool pressure has a negligible affect upon the specific volume of the water, pool pressure is taken as atmospheric.

Intermediate NPSHr
Between 7-20 hrs, the NPSHr for 2 RHR pump operation increases from 23.6' to 29.0'. Conservatively ignoring this ramp up, the NPSHr at 7 hrs (25,200 seconds) is taken as 29.0'. As can be inferred from Table 4.3 "NPSHa vs. decreasing pool temperature" the pool only needs to cool about 13°F to 162°F over 17,200 seconds, to achieve an adequate NPSHa of about 29.0' without crediting any overpressure. This equates to a cooldown rate of about 0.8°F/1000 seconds, which is much less than the exhibited cooldown rate between 3000-8000 seconds. (Note that if the 20 hour (72,000 sec) time is used, then the required cooldown rate becomes 13°F over 64,000 sec or about 0.2°F/1000 sec.)

In consideration of the conservative selection of NPSHr (@ 7 hrs) and the minimal cooldown rate required, there is adequate NPSHa without crediting overpressure.

Maximum NPSHr

Between 20-100 hrs, the NPSHr for 2 RHR pump operation increases from 29.0' to 30.0'. Conservatively ignoring this ramp up, the NPSHr at 20 hrs (72,000 seconds) is taken as 30.0'. As can be inferred from Table 4.3 "NPSHa vs. decreasing pool temperature" the pool only needs to cool about 4°F to 158°F between 7hrs and 20 hrs (46,800 seconds), to achieve an adequate NPSHa of about 30.0' without crediting any overpressure. This equates to a cooldown rate of about 0.1°F/1000 seconds.

In consideration of the conservative selection of NPSHr (@ 20 hrs) and the minimal cooldown rate required, there is adequate NPSHa without crediting overpressure.

Suppression Pool Cooldown Comparison to LOCA

The temperature profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-177 (Ref: 1). The pool temperature rises until about 24,000 seconds (~7 hrs).

Shortly after peaking, there is essentially a uniform decrease in temperature between ~9hrs (194.1°F) and 30hrs (174.4°F) of about 0.2°F/1000 sec. Between ~30hrs (174.4°F) and the end of the tabulated data at ~48hrs (166.2°F), the uniform decrease in temperature is about 0.1°F/1000sec.

The required suppression pool cooldown rates, projected above for ATWS, are similar to the suppression pool cooldown rates developed for LOCA.

4.4 General Profile – Overpressure Required vs Pool Temperature

A wide temperature range is evaluated, up to 205°F, which is about 10°F more than the peak LOCA temperature addressed by GE-VYNPS-AEP-177 (Ref: 1).

The NPSH evaluation of general overpressure requirements for the RHR and CS pumps is documented in Table 4.4 for the minimum NPSHr (0-7 hours of operation) and maximum NPSHr (>7 hours of operation). The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results. Further discussion of selected terms is presented below.

Flow rates – Q (gpm) - RHR

1 RHR: 7400 and 7000*gpm
2 RHR: 14,200 gpm (7100 gpm each) and 12,800 gpm (6400* gpm each)
* these values are conveniently selected, based upon the pump vendor's data, for the purpose of establishing a profile range.

Flow rates – Q (gpm) - CS

CS: 4600 gpm and 3500 gpm

Suction Elevation Head, Z

Based on a minimum suppression pool volume of 68,000 ft^3.

RHR: 11.25' as calculated in Section 4.3

CS: 11.42' based on the evaluation documented in Section 4.3 of this CCN and adjusting for the CS short term Z (12.47'). Therefore the adjusted Z is 12.47'-1.05' = 11.42'.

Clean Strainer Losses (hs)

Refer to Section 3.6 of VYC-0808 (Ref: 4)

1 RHR @ 7400 gpm = 0.33'

1 RHR @ 7000 gpm = 0.30' = 0.33*(Q/7400)^2**

2 RHR @ 14,200 gpm (7100 each) = 1.22'

2 RHR @ 12,800 gpm (6400 each) = 0.99' = 1.22*(Q/14200)^2**

** head loss at other than the reference flow rate is proportional to the square of the flow ratio

CS @ 4600 gpm = 0.51'

CS @ 3500 gpm = 0.29 = 0.38*(Q/4000)^2

Maximum Debris Losses (hd)

This term is not applicable since no high-energy line break (HELB) is postulated to dislodge insulation and create debris in the suppression pool.

Minimum NPRHr

1 RHR @ 7400 gpm = 23.8' Refer to Section 4.1

1 RHR @ 7000 gpm = 23.5' Refer to Section 4.1

2 RHR @ 7100 gpm (each) = 23.6' Refer to Section 4.1

2 RHR @ 6400 gpm (each) = 23.0' (see below)

Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 6400 gpm an allowable NPSH of 23.0 ft is acceptable for less than 7 hrs of operation.

CS @ 4600 gpm = 28.0' Refer to Section 4.1

CS @ 3500 gpm = 24.8' (see below)

Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 3500 gpm an allowable NPSH of 24.8 ft is acceptable for less than 7 hrs of operation.

Maximum NPRHr
1 RHR @ 7400 gpm = 31.7' Refer to Section 4.2
1 RHR @ 7000 gpm = 29.5' Refer to Section 4.3

2 RHR @ 7100 gpm (each) = 30.0' Refer to Section 4.3
2 RHR @ 6400 gpm (each) = 28.5' (see below)

Figure 2.1-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 6400 gpm an allowable NPSH of 28.5 ft is acceptable at greater than 100 hrs of operation.

CS @ 4600 gpm = 35.0' (see below)
CS @ 3500 gpm = 29.6' Refer to Section 4.2

Figure 2.2-1 of Attachment 3 to calculation VYC-0808 provides a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates. This plot shows that at 4600 gpm an allowable NPSH of 35.0 ft is acceptable at greater than 100 hrs of operation.

5.0 Summary of Results

NPSHa is rounded to the nearest 0.1ft and OPR, OPC, and OPA are rounded to the nearest 0.1psig.

5.1 LOCA - Short Term (0-600 sec):

NPSHa is adequate for both CS and RHR pumps without crediting overpressure. NPSHa shown below is at the peak temperature.

Pump	Total flow, gpm	NPSHr, ft	NPSHa, ft
CS	4,600	28.0	28.4
1RHR	7,400	23.8	31.1
2 RHR	14,200	23.6	28.8

5.2 LOCA - Long Term (>600 sec):

NPSHa is adequate for both CS and RHR pumps with an overpressure credit that varies over time, as shown in Fig. 4.2. NPSHa, OPR, OPC, OPA are shown below, at the peak temperature

Pump	Total flow, gpm	NPSHr, ft	NPSHa, ft	OPR, psig	OPC, psig	OPA, psig
CS	3,500	29.6	19.5	4.2	6.1	8.1
1RHR	7,400	31.7	19.6	5.1	6.1	8.1

5.3 ATWS – (<7hrs):

NPSHa is adequate for RHR pumps with an overpressure credit of 2.4 psig required between 600-6000 seconds, as shown in Fig. 4.3. NPSHa, OPR, OPC, OPA are shown below, at the peak temperature.

Pump	Total flow, gpm	NPSHr, ft	NPSHa, ft	OPR, psig	OPC, psig	OPA, psig
1RHR	7,400	23.8	21.1	1.1	2.4	12.3
2RHR	14,200	23.6	18.9	2.0	2.4	12.3

Note that for the same time period, this overpressure credit (2.4 psig) is bounded by that required for LOCA Long Term (2.4 to 4.4 psig).

5.4 ATWS – (>7hrs):

NPSHa is adequate for the RHR pumps without crediting overpressure. This is due to the significant rate of reduction of the suppression pool temperature over time. The reduction in temperature, and associated vapor pressure, will more than offset increases in NPHr at extended operating times. Note that between 3000 –8000 seconds (the end of tabulated data provided by GE) the pool temperature decreases to 175°F, at the substantial uniform rate of about 2.4°F/1000sec. The increased NPSHr, at significant time intervals, is shown below along with the approximate pool temp, in parentheses, at or below which, over pressure credit is not required.

Pump	Total flow, gpm	0-7 hr NPSHr, ft	20 hr NPSHr, ft	>100 hr NPSHr, ft
1RHR	7,400	23.8	30.5 (164°F)	31.7 (160°F)
2RHR	14,200	23.6	29.0 (162 °F)	30.0 (158°F)

5.5 General Profile – Overpressure Required vs. Pool Temperature

General profiles of "Overpressure Required vs. Pool Temperature", for the scenarios listed below are provided in Figures 4.4-1 through 4.4-4. The profiles are intended to serve as the basis for determining overpressure requirements when performing RHR and CS pump NPSH evaluation for any other events, which cause elevated suppression pool temperatures, without strainer debris loading. A representative flow range is presented based on available vendor data for NPSH.

Profiles

Figure	Pumps operating	Flow range, gpm	NPSHr
4.4-1	1 RHR	7000-7400	Minimum (0-7 hrs of operation)
4.4-1	1 RHR	7000-7400	Maximum (>7 hrs of operation)
4.4-2	2 RHR	12,800-14,200	Minimum (0-7 hrs of operation)
4.4-2	2 RHR	12,800-14,200	Maximum (>7 hrs of operation)
4.4-3	1 CS	3500-4600	Minimum (0-7 hrs of operation)
4.4-4	1 CS	3500-4600	Maximum (>7 hrs of operation)

6.0 Conclusions

There is adequate NPSH available for operating the RHR and CS pumps at EPU conditions for the DBA-Loss of Coolant Accident (LOCA), short term, without crediting torus overpressure.

Torus overpressure must be credited for operating the RHR and CS pumps at EPU conditions for the following events in order to achieve adequate NPSH available:

- DBA-Loss of Coolant Accident (LOCA), long term
- Anticipated Transients Without Scram (ATWS)

The overpressure credit required for LOCA bounds that required for ATWS.

A basis for readily determining overpressure requirements, when performing RHR and CS pump NPSH evaluation for any other events which cause elevated suppression pool temperatures has provided in the form of a family of curves profiling overpressure required vs. pool temperature.

The results of this CCN will provide input to the PUSAR (Ref: 12) for the RHR and CS NPSH evaluation and will alter input to calculation VYC-1628 (Ref: 13) to address the need for crediting torus overpressure in the calculation of NPSH available. Note that calculation VYC-1628 may be superseded by GE EPU Analysis. The need for crediting torus overpressure in the RHR and CS NPSH evaluation, shall also be addressed in the SADBD (Ref: 14), UFSAR (Ref: 15), and system DBDs RHR (Ref: 16) and CS (Ref: 17).

Note that use of overpressure credit must be approved by the NRC as part of EPU.

No specific 50.59 Screening/Evaluation is required for this CCN since all EPU design changes and associated 50.59 documentation will be part of VYDC-2003-008.

Table 4.1
LOCA - Short term

LOCA - Short Term

NPSHa = (14.7-Pg)(144Vf)+Z-hf-hs-hd
OPR = (NPSHr – NPSHa)/(144*Vf)
OPA = Over pressure available
OPC = Over pressure credited

Cross references:
Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
See Discussion in Section 4.0 of this CCN

Short Term Flow Rate (gpm)
1 RHR Q = 7400 CS Q = 4600 — Table of 1 calc VYC-0808 Rev 6 (Ref: 4)
2 RHR Q = 14200

Suction Line Losses (ft)
1 RHR hf = 4.77E-8*Q^2 CS hf = 2.5E-7*Q^2 — Section 3.7 of VYC-0808 Rev 6 (Ref: 4)
2 RHR hf = 7.84E-8*(Q/2)^2

Clean Strainer Losses (ft)
1 RHR hs = 0.33 CS hs = 0.51 — Section 3.6 of VYC-0808 Rev 6 (Ref: 4)
2 RHR hs = 1.22

Maximum Debris Losses (ft) @ >=base temperature — See discussion in Section 4.1 of this CCN
1 RHR hd = 0.33@173F CS hd = 0.32@173F — Ref: 2, 3, 4
2 RHR hd = 0.48@170F — Ref: 2, 4

Maximum Debris Losses (ft) @ < base temperature
1 RHR hd = .33*(173/T) CS hd = .32*(173/T) — See discussion in Section 2.0 of this CCN
2 RHR hd = .48*(170/T)
where T = suppression pool temperature, F

Elevation Head (ft)
RHR Z = 12.3 CS Z = 12.47 — See discussion in Section 4.1 of this CCN for conservatism.

NPSHr (ft)
1 RHR NPSHr = 23.8 CS NPSHr = 28.0 — See discussion in Section 4.1 of this CCN
2 RHR NPSHr = 23.6

Short Term (After EPU) - Peak Torus Temperature

Pump(s)	Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)	OPA (psig)	OPC (psig)
CS	600	165.1	17.75	5.349	0.016423	12.47	5.29	0.51	0.34	28.44	28.00	0.00	3.05	0.00
1 RHR	600	165.1	17.75	5.349	0.016423	12.30	2.61	0.33	0.35	31.12	23.80	0.00	3.05	0.00
2 RHR	600	165.1	17.75	5.349	0.016423	12.30	3.95	1.22	0.49	28.75	23.60	0.00	3.05	0.00

Table 4.2
LOCA - Long term

LOCA - Long Term

NPSHa =	(14.7-Pg)(144Vf)+Z-hf-hs-hd	
OPR =	Over prossure required (NPSHr – NPSHa)/(144*Vf)	
OPA =	Over pressure available	
OPC =	Over pressure credited	

Cross references:
Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
See Discussion in Section 4.1 of this CCN

Long Term Flow Rate (gpm)				
1 RHR	Q = 7400	CS	Q = 3500	Table of 1 calc VYC-0808 Rev 6 (Ref: 4)
Suction Line Losses (ft)				
1 RHR	hf = 4.77E-8*Q^2	CS	hf = 2.5E-7*Q^2	Section 3.7 of VYC-0808 Rev 6 (Ref: 4)
Clean Strainer Losses (ft)				
1 RHR	hs = 0.33	CS	hs = .38*(Q/4000)^2 for Q<= 4000	Section 3.6 of VYC-0808 Rev 6 (Ref: 4)
Maximum Debris Losses (ft) @ >= 173F				
1 RHR	hd = 0.33	CS	hd = 0.21	See discussion in Section 4.2 of this CCN
Maximum Debris Losses (ft) @ < 181.7F				
1 RHR	hd = .33*(173/T)	CS	hd = .21*(173/T)	See discussion in Section 2.0 of this CCN
where T = suppression pool temperature, F				
Elevation Head (ft)				
RHR	Z = 12.4	CS	Z = 12.57	See discussion in Section 4.2 of this CCN for conservatism.
NPSHr (ft)				
1 RHR	NPSHr = 31.7	CS	NPSHr = 29.6	See discussion in Section 4.2 of this CCN

Table 4.2
LOCA - Long term

CS - Long Term (After EPU)

Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	CS NPSHa (ft)	CS NPSHr (ft)	CS OPR (psig)	OPA (psig)	OPC (psig)
779	169.6	17.88	5.938	0.016448	12.57	3.06	0.29	0.21	29.76	29.60	0.00	3.18	2.40
1,400	173.6	18.34	6.506	0.016471	12.57	3.06	0.29	0.21	28.44	29.60	0.49	3.64	2.40
1,711	175.2	18.51	6.746	0.016481	12.57	3.06	0.29	0.21	27.88	29.60	0.72	3.81	2.40
2,022	176.6	18.72	6.962	0.016489	12.57	3.06	0.29	0.21	27.38	29.60	0.94	4.02	3.40
2,951	180.0	19.30	7.511	0.016509	12.57	3.06	0.29	0.21	26.10	29.60	1.47	4.60	3.40
3,876	182.6	19.90	7.955	0.016525	12.57	3.06	0.29	0.21	25.06	29.60	1.91	5.20	3.40
4,185	183.3	20.07	8.078	0.016529	12.57	3.06	0.29	0.21	24.77	29.60	2.03	5.37	4.40
5,109	185.1	20.51	8.402	0.016541	12.57	3.06	0.29	0.21	24.01	29.60	2.35	5.81	4.40
5,997	186.6	20.93	8.680	0.016550	12.57	3.06	0.29	0.21	23.35	29.60	2.62	6.23	4.40
7,121	188.1	21.42	8.966	0.016559	12.57	3.06	0.29	0.21	22.68	29.60	2.90	6.72	5.10
8,035	189.1	21.73	9.161	0.016566	12.57	3.06	0.29	0.21	22.22	29.60	3.09	7.03	5.10
8,970	190.0	21.92	9.340	0.016571	12.57	3.06	0.29	0.21	21.80	29.60	3.27	7.22	5.10
10,191	191.0	22.10	9.541	0.016578	12.57	3.06	0.29	0.21	21.32	29.60	3.47	7.40	6.10
12,040	192.1	22.31	9.767	0.016585	12.57	3.06	0.29	0.21	20.79	29.60	3.69	7.61	6.10
13,915	193.1	22.47	9.977	0.016591	12.57	3.06	0.29	0.21	20.29	29.60	3.90	7.77	6.10
16,375	194.0	22.63	10.168	0.016597	12.57	3.06	0.29	0.21	19.84	29.60	4.08	7.93	6.10
20,064	194.6	22.74	10.298	0.016601	12.57	3.06	0.29	0.21	19.53	29.60	4.21	8.04	6.10
24,094	194.7	22.77	10.320	0.016601	12.57	3.06	0.29	0.21	19.48	29.60	4.23	8.07	6.10
30,391	194.4	22.73	10.255	0.016599	12.57	3.06	0.29	0.21	19.63	29.60	4.17	8.03	6.10
40,033	192.5	22.36	9.851	0.016587	12.57	3.06	0.29	0.21	20.59	29.60	3.77	7.66	5.60
51,025	189.5	21.77	9.240	0.016568	12.57	3.06	0.29	0.21	22.03	29.60	3.17	7.07	5.10
61,023	186.7	21.29	8.699	0.016550	12.57	3.06	0.29	0.21	23.31	29.60	2.64	6.59	4.60
70,699	183.9	20.82	8.185	0.016533	12.57	3.06	0.29	0.21	24.52	29.60	2.13	6.12	4.10
80,699	181.3	20.40	7.730	0.016517	12.57	3.06	0.29	0.21	25.58	29.60	1.69	5.70	3.60
90,697	178.8	20.03	7.313	0.016502	12.57	3.06	0.29	0.21	26.56	29.60	1.28	5.33	3.10
100,697	176.4	19.68	6.931	0.016488	12.57	3.06	0.29	0.21	27.45	29.60	0.90	4.98	3.10
110,697	174.4	19.39	6.625	0.016476	12.57	3.06	0.29	0.21	28.16	29.60	0.60	4.69	2.60
120,697	172.9	19.14	6.404	0.016467	12.57	3.06	0.29	0.21	28.68	29.60	0.39	4.44	2.60
130,685	171.5	18.92	6.202	0.016459	12.57	3.06	0.29	0.21	29.15	29.60	0.19	4.22	2.10
140,685	170.2	18.72	6.020	0.016452	12.57	3.06	0.29	0.21	29.57	29.60	0.01	4.02	2.10
150,685	169.0	18.53	5.856	0.016445	12.57	3.06	0.29	0.21	29.95	29.60	0.00	3.83	1.70
160,685	167.7	18.35	5.683	0.016437	12.57	3.06	0.29	0.22	30.34	29.60	0.00	3.65	1.70
170,685	166.5	18.20	5.526	0.016431	12.57	3.06	0.29	0.22	30.70	29.60	0.00	3.50	1.30
172,800	166.2	18.16	5.488	0.016429	12.57	3.06	0.29	0.22	30.79	29.60	0.00	3.46	1.30

Table 4.2
LOCA - Long term

RHR - Long Term (After EPU)

Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	RHR NPSHa (ft)	RHR NPSHr (ft)	RHR OPR (psig)	OPA (psig)	OPC (psig)
779	169.6	17.88	5.938	0.016448	12.40	2.61	0.33	0.34	29.87	31.70	0.77	3.18	2.40
1,400	173.6	18.34	6.506	0.016471	12.40	2.61	0.33	0.33	28.56	31.70	1.32	3.64	2.40
1,711	175.2	18.51	6.746	0.016481	12.40	2.61	0.33	0.33	28.00	31.70	1.56	3.81	2.40
2,022	176.6	18.72	6.962	0.016489	12.40	2.61	0.33	0.33	27.50	31.70	1.77	4.02	3.40
2,951	180.0	19.30	7.511	0.016509	12.40	2.61	0.33	0.33	26.22	31.70	2.31	4.60	3.40
3,876	182.6	19.90	7.955	0.016525	12.40	2.61	0.33	0.33	25.18	31.70	2.74	5.20	3.40
4,185	183.3	20.07	8.078	0.016529	12.40	2.61	0.33	0.33	24.89	31.70	2.86	5.37	4.40
5,109	185.1	20.51	8.402	0.016541	12.40	2.61	0.33	0.33	24.13	31.70	3.18	5.81	4.40
5,997	186.6	20.93	8.680	0.016550	12.40	2.61	0.33	0.33	23.47	31.70	3.45	6.23	4.40
7,121	188.1	21.42	8.966	0.016559	12.40	2.61	0.33	0.33	22.80	31.70	3.73	6.72	5.10
8,035	189.1	21.73	9.161	0.016566	12.40	2.61	0.33	0.33	22.34	31.70	3.92	7.03	5.10
8,970	190.0	21.92	9.340	0.016571	12.40	2.61	0.33	0.33	21.92	31.70	4.10	7.22	5.10
10,191	191.0	22.10	9.541	0.016578	12.40	2.61	0.33	0.33	21.44	31.70	4.30	7.40	6.10
12,040	192.1	22.31	9.767	0.016585	12.40	2.61	0.33	0.33	20.91	31.70	4.52	7.61	6.10
13,915	193.1	22.47	9.977	0.016591	12.40	2.61	0.33	0.33	20.41	31.70	4.72	7.77	6.10
16,375	194.0	22.63	10.168	0.016597	12.40	2.61	0.33	0.33	19.96	31.70	4.91	7.93	6.10
20,064	194.6	22.74	10.298	0.016601	12.40	2.61	0.33	0.33	19.65	31.70	5.04	8.04	6.10
24,094	194.7	22.77	10.320	0.016601	12.40	2.61	0.33	0.33	19.60	31.70	5.06	8.07	6.10
30,391	194.4	22.73	10.255	0.016599	12.40	2.61	0.33	0.33	19.75	31.70	5.00	8.03	6.10
40,033	192.5	22.36	9.851	0.016587	12.40	2.61	0.33	0.33	20.71	31.70	4.60	7.66	5.60
51,025	189.5	21.77	9.240	0.016568	12.40	2.61	0.33	0.33	22.15	31.70	4.00	7.07	5.10
61,023	186.7	21.29	8.699	0.016550	12.40	2.61	0.33	0.33	23.43	31.70	3.47	6.59	4.60
70,699	183.9	20.82	8.185	0.016533	12.40	2.61	0.33	0.33	24.64	31.70	2.97	6.12	4.10
80,699	181.3	20.40	7.730	0.016517	12.40	2.61	0.33	0.33	25.71	31.70	2.52	5.70	3.60
90,697	178.8	20.03	7.313	0.016502	12.40	2.61	0.33	0.33	26.68	31.70	2.11	5.33	3.10
100,697	176.4	19.68	6.931	0.016488	12.40	2.61	0.33	0.33	27.57	31.70	1.74	4.98	3.10
110,697	174.4	19.39	6.625	0.016476	12.40	2.61	0.33	0.33	28.29	31.70	1.44	4.69	2.60
120,697	172.9	19.14	6.404	0.016467	12.40	2.61	0.33	0.33	28.80	31.70	1.22	4.44	2.60
130,685	171.5	18.92	6.202	0.016459	12.40	2.61	0.33	0.33	29.27	31.70	1.03	4.22	2.10
140,685	170.2	18.72	6.020	0.016452	12.40	2.61	0.33	0.34	29.68	31.70	0.85	4.02	2.10
150,685	169.0	18.53	5.856	0.016445	12.40	2.61	0.33	0.34	30.06	31.70	0.69	3.83	1.70
160,685	167.7	18.35	5.683	0.016437	12.40	2.61	0.33	0.34	30.46	31.70	0.52	3.65	1.70
170,685	166.5	18.20	5.526	0.016431	12.40	2.61	0.33	0.34	30.82	31.70	0.37	3.50	1.30
172,800	166.2	18.16	5.488	0.016429	12.40	2.61	0.33	0.34	30.91	31.70	0.33	3.46	1.30

Table 4.3
ATWS

ATWS

NPSHa =	(14.7-Pg)(144Vf)+Z-hf-hs-hd		
OPR =	(NPSHr – NPSHa)/(144*Vf)		
OPA =	Over pressure available		
OPC =	Over pressure credited		
NPSHcm =	NPSH credited margin (NPSHa+OPC(144*vf) -NPSHr)		

Cross references:
Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
See Discussion in Section 4.0 of this CCN

							Cross references
Flow Rate (gpm)							
1 RHR	Q =	7400	2 RHR	Q =	14200		Table of 1 calc VYC-0808 Rev 6 (Ref: 4)
Suction Line Losses (ft)							
1 RHR	hf =	4.77E-8*Q^2	2 RHR	hf =	7.84E-8*(Q/2)^2		Section 3.7 of VYC-0808 Rev 6 (Ref: 4)
Clean Strainer Losses (ft)							
1 RHR	hs =	0.33	2 RHR	hs =	1.22		Section 3.6 of VYC-0808 Rev 6 (Ref: 4)
Maximum Debris Losses (ft)							See discussion in Section 4.3 of this CCN
1 RHR	hd =	0	2 RHR	hd =	0		
Elevation Head (ft)							
RHR	Z =	11.25					See discussion in Section 4.3 of this CCN

NPSHr (ft)		0-7 hrs	>7 hrs	>20 hrs	
1 RHR	NPSHr =	23.8	30.5	31.7	See discussion in Section 4.3 of this CCN
2 RHR	NPSHr =	23.6	29.0	30.0	

Table 4.3
ATWS

Minimum NPSHr (0-7 hrs of operation)

Pump(s)	Time (sec)	GE Pool Temp (F)	GE Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)	OPA (psig)	OPC (psig)
1 RHR	300	160.0	21.00	4.741	0.016394	11.25	2.61	0.33	0.00	31.82	23.80	-3.40	6.30	0.00
1 RHR	600	175.0	22.90	6.716	0.016479	11.25	2.61	0.33	0.00	27.25	23.80	-1.46	8.20	0.00
1 RHR	1,000	182.0	25.40	7.850	0.016521	11.25	2.61	0.33	0.00	24.60	23.80	-0.34	10.70	2.40
1 RHR	1,300	187.0	26.20	8.756	0.016552	11.25	2.61	0.33	0.00	22.48	23.80	0.56	11.50	2.40
1 RHR	1,838	190.0	27.00	9.340	0.016571	11.25	2.61	0.33	0.00	21.10	23.80	1.13	12.30	2.40
1 RHR	3,000	187.0	26.60	8.756	0.016552	11.25	2.61	0.33	0.00	22.48	23.80	0.56	11.90	2.40
1 RHR	5,000	182.0	25.90	7.850	0.016521	11.25	2.61	0.33	0.00	24.60	23.80	-0.34	11.20	2.40
1 RHR	6,000	180.0	25.50	7.511	0.016509	11.25	2.61	0.33	0.00	25.40	23.80	-0.67	10.80	0.00
1 RHR	8,000	175.0	24.70	6.716	0.016479	11.25	2.61	0.33	0.00	27.25	23.80	-1.46	10.00	0.00
2 RHR	300	160.0	21.00	4.741	0.016394	11.25	3.95	1.22	0.00	29.59	23.60	-2.54	6.30	0.00
2 RHR	600	175.0	22.90	6.716	0.016479	11.25	3.95	1.22	0.00	25.02	23.60	-0.60	8.20	2.40
2 RHR	1,000	182.0	25.40	7.850	0.016521	11.25	3.95	1.22	0.00	22.37	23.60	0.52	10.70	2.40
2 RHR	1,300	187.0	26.20	8.756	0.016552	11.25	3.95	1.22	0.00	20.25	23.60	1.41	11.50	2.40
2 RHR	1,838	190.0	27.00	9.340	0.016571	11.25	3.95	1.22	0.00	18.87	23.60	1.98	12.30	2.40
2 RHR	3,000	187.0	26.60	8.756	0.016552	11.25	3.95	1.22	0.00	20.25	23.60	1.41	11.90	2.40
2 RHR	5,000	182.0	25.90	7.850	0.016521	11.25	3.95	1.22	0.00	22.37	23.60	0.52	11.20	2.40
2 RHR	6,000	180.0	25.50	7.511	0.016509	11.25	3.95	1.22	0.00	23.17	23.60	0.18	10.80	2.40
2 RHR	8,000	175.0	24.70	6.716	0.016479	11.25	3.95	1.22	0.00	25.02	23.60	-0.60	10.00	0.00

NPSHa vs decreasing pool temperature

1 RHR		166.0	14.70	5.462	0.016428	11.25	2.61	0.33	0.00	30.16				
1 RHR		164.0	14.70	5.212	0.016417	11.25	2.61	0.33	0.00	30.74				
1 RHR		162.0	14.70	4.972	0.016406	11.25	2.61	0.33	0.00	31.29				
1 RHR		160.0	14.70	4.741	0.016395	11.25	2.61	0.33	0.00	31.82				
1 RHR		158.0	14.70	4.520	0.016384	11.25	2.61	0.33	0.00	32.33				
1 RHR		156.0	14.70	4.307	0.016373	11.25	2.61	0.33	0.00	32.81				
2 RHR		166.0	14.70	5.462	0.016428	11.25	3.95	1.22	0.00	27.93				
2 RHR		164.0	14.70	5.212	0.016417	11.25	3.95	1.22	0.00	28.51				
2 RHR		162.0	14.70	4.972	0.016406	11.25	3.95	1.22	0.00	29.06				
2 RHR		160.0	14.70	4.741	0.016395	11.25	3.95	1.22	0.00	29.59				
2 RHR		158.0	14.70	4.520	0.016384	11.25	3.95	1.22	0.00	30.10				
2 RHR		156.0	14.70	4.307	0.016373	11.25	3.95	1.22	0.00	30.58				

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature

NPSHa = (14.7-Pg)(144Vf)+Z-hf-hs-hd
OPR = (NPSHr – NPSHa)/(144*Vf)

Cross references:
Section 2.3 of VYC-0808 Rev 6 (Ref: 4)
See Discussion in Section 4.0 and 3.0 of this CCN

Flow Rate (gpm)

	1 RHR	2 RHR	CS	Cross reference
high Q	7400	14200	4600	See discussion in Section 4.4 of this CCN
low Q	7000	12800	3500	

Suction Line Losses (ft)

	1 RHR	2 RHR	CS	Cross reference
hf =	4.77E-8*Q^2	7.84E-8*(Q/2)^2	2.5E-7*Q^2	Section 3.7 of VYC-0808 Rev 6 (Ref: 4)

Clean Strainer Losses (ft) @ flow rates tabulated above

	1 RHR	2 RHR	CS	Cross reference
(high Q) hs	0.33	1.22	0.51	See discussion in Section 4.4 of this CCN
(low Q) hs	.33*(Q/7400)^2	1.22*(Q/14200)^2	.38*(Q/4000)^2	

Maximum Debris Losses (ft)

	1 RHR	2 RHR	CS	Cross reference
hd	0	0	0	See discussion in Section 4.4 of this CCN

Elevation Head (ft)

	RHR	CS	Cross reference
Z	11.25	11.42	See discussion in Section 4.4 of this CCN

Minimum NPSHr (ft) @ flow rates tabulated above

	1 RHR	2 RHR	CS	Cross reference
(High Q) NPSHr	23.8	23.6	28.0	See discussion in Section 4.4 of this CCN
(Low Q) NPSHr	23.5	23.0	24.8	

Maximum NPSHr (ft) @ flow rates tabulated above

	1 RHR	2 RHR	CS	Cross reference
(High Q) NPSHr	31.7	30.0	35.0	See discussion in Section 4.4 of this CCN
(Low Q) NPSHr	29.5	28.5	29.6	

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature **1 RHR @ Min NPSHr** **(0-7 hrs of operation)**

Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
1RHR	7,400	175.0	14.70	6.716	0.016480	11.25	2.61	0.33	0.00	27.25	23.80	-1.46
1RHR	7,400	180.0	14.70	7.511	0.016510	11.25	2.61	0.33	0.00	25.40	23.80	-0.67
1RHR	7,400	185.0	14.70	8.384	0.016540	11.25	2.61	0.33	0.00	23.35	23.80	0.19
1RHR	7,400	190.0	14.70	9.340	0.016572	11.25	2.61	0.33	0.00	21.10	23.80	1.13
1RHR	7,400	195.0	14.70	10.385	0.016604	11.25	2.61	0.33	0.00	18.63	23.80	2.16
1RHR	7,400	200.0	14.70	11.526	0.016637	11.25	2.61	0.33	0.00	15.91	23.80	3.29
1RHR	7,400	205.0	14.70	12.770	0.016670	11.25	2.61	0.33	0.00	12.94	23.80	4.52
1RHR	7,000	175.0	14.70	6.716	0.016480	11.25	2.34	0.30	0.00	27.56	23.50	-1.71
1RHR	7,000	180.0	14.70	7.511	0.016510	11.25	2.34	0.30	0.00	25.70	23.50	-0.93
1RHR	7,000	185.0	14.70	8.384	0.016540	11.25	2.34	0.30	0.00	23.66	23.50	-0.07
1RHR	7,000	190.0	14.70	9.340	0.016572	11.25	2.34	0.30	0.00	21.40	23.50	0.88
1RHR	7,000	195.0	14.70	10.385	0.016604	11.25	2.34	0.30	0.00	18.93	23.50	1.91
1RHR	7,000	200.0	14.70	11.526	0.016637	11.25	2.34	0.30	0.00	16.22	23.50	3.04
1RHR	7,000	205.0	14.70	12.770	0.016670	11.25	2.34	0.30	0.00	13.25	23.50	4.27

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature **2 RHR @ Min NPSHr** **(0-7 hrs of operation)**

Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
2RHR	14,200	170.0	14.70	5.993	0.016451	11.25	3.95	1.22	0.00	26.70	23.60	-1.31
2RHR	14,200	175.0	14.70	6.716	0.016480	11.25	3.95	1.22	0.00	25.02	23.60	-0.60
2RHR	14,200	180.0	14.70	7.511	0.016510	11.25	3.95	1.22	0.00	23.17	23.60	0.18
2RHR	14,200	185.0	14.70	8.384	0.016540	11.25	3.95	1.22	0.00	21.12	23.60	1.04
2RHR	14,200	190.0	14.70	9.340	0.016572	11.25	3.95	1.22	0.00	18.87	23.60	1.98
2RHR	14,200	195.0	14.70	10.385	0.016604	11.25	3.95	1.22	0.00	16.39	23.60	3.01
2RHR	14,200	200.0	14.70	11.526	0.016637	11.25	3.95	1.22	0.00	13.68	23.60	4.14
2RHR	14,200	205.0	14.70	12.770	0.016670	11.25	3.95	1.22	0.00	10.71	23.60	5.37
2RHR	12,800	170.0	14.70	5.993	0.016451	11.25	3.21	0.99	0.00	27.68	23.00	-1.97
2RHR	12,800	175.0	14.70	6.716	0.016480	11.25	3.21	0.99	0.00	26.00	23.00	-1.26
2RHR	12,800	180.0	14.70	7.511	0.016510	11.25	3.21	0.99	0.00	24.14	23.00	-0.48
2RHR	12,800	185.0	14.70	8.384	0.016540	11.25	3.21	0.99	0.00	22.09	23.00	0.38
2RHR	12,800	190.0	14.70	9.340	0.016572	11.25	3.21	0.99	0.00	19.84	23.00	1.32
2RHR	12,800	195.0	14.70	10.385	0.016604	11.25	3.21	0.99	0.00	17.37	23.00	2.36
2RHR	12,800	200.0	14.70	11.526	0.016637	11.25	3.21	0.99	0.00	14.65	23.00	3.48
2RHR	12,800	205.0	14.70	12.770	0.016670	11.25	3.21	0.99	0.00	11.68	23.00	4.72

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature **CS Min NPSHr** **(0-7 hrs of operation)**

Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
CS	4,600	155.0	14.70	4.204	0.016368	11.42	5.29	0.51	0.00	30.36	28.00	-1.00
CS	4,600	160.0	14.70	4.741	0.016395	11.42	5.29	0.51	0.00	29.13	28.00	-0.48
CS	4,600	165.0	14.70	5.336	0.016422	11.42	5.29	0.51	0.00	27.76	28.00	0.10
CS	4,600	170.0	14.70	5.993	0.016451	11.42	5.29	0.51	0.00	26.25	28.00	0.74
CS	4,600	175.0	14.70	6.716	0.016480	11.42	5.29	0.51	0.00	24.57	28.00	1.45
CS	4,600	180.0	14.70	7.511	0.016510	11.42	5.29	0.51	0.00	22.71	28.00	2.22
CS	4,600	185.0	14.70	8.384	0.016540	11.42	5.29	0.51	0.00	20.66	28.00	3.08
CS	4,600	190.0	14.70	9.340	0.016572	11.42	5.29	0.51	0.00	18.41	28.00	4.02
CS	4,600	195.0	14.70	10.385	0.016604	11.42	5.29	0.51	0.00	15.94	28.00	5.05
CS	4,600	200.0	14.70	11.526	0.016637	11.42	5.29	0.51	0.00	13.22	28.00	6.17
CS	4,600	205.0	14.70	12.770	0.016670	11.42	5.29	0.51	0.00	10.25	28.00	7.39
CS	3,500	155.0	14.70	4.204	0.016368	11.42	3.06	0.29	0.00	32.81	24.80	-3.40
CS	3,500	160.0	14.70	4.741	0.016395	11.42	3.06	0.29	0.00	31.58	24.80	-2.87
CS	3,500	165.0	14.70	5.336	0.016422	11.42	3.06	0.29	0.00	30.21	24.80	-2.29
CS	3,500	170.0	14.70	5.993	0.016451	11.42	3.06	0.29	0.00	28.69	24.80	-1.64
CS	3,500	175.0	14.70	6.716	0.016480	11.42	3.06	0.29	0.00	27.01	24.80	-0.93
CS	3,500	180.0	14.70	7.511	0.016510	11.42	3.06	0.29	0.00	25.16	24.80	-0.15
CS	3,500	185.0	14.70	8.384	0.016540	11.42	3.06	0.29	0.00	23.11	24.80	0.71
CS	3,500	190.0	14.70	9.340	0.016572	11.42	3.06	0.29	0.00	20.86	24.80	1.65
CS	3,500	195.0	14.70	10.385	0.016604	11.42	3.06	0.29	0.00	18.38	24.80	2.68
CS	3,500	200.0	14.70	11.526	0.016637	11.42	3.06	0.29	0.00	15.67	24.80	3.81
CS	3,500	205.0	14.70	12.770	0.016670	11.42	3.06	0.29	0.00	12.70	24.80	5.04

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature **1 RHR @ Max NPSHr** **(>7 hrs of operation)**

Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
1RHR	7,400	155.0	14.70	4.204	0.016368	11.25	2.61	0.33	0.00	33.05	31.70	-0.57
1RHR	7,400	160.0	14.70	4.741	0.016395	11.25	2.61	0.33	0.00	31.82	31.70	-0.05
1RHR	7,400	165.0	14.70	5.336	0.016422	11.25	2.61	0.33	0.00	30.45	31.70	0.53
1RHR	7,400	170.0	14.70	5.993	0.016451	11.25	2.61	0.33	0.00	28.93	31.70	1.17
1RHR	7,400	175.0	14.70	6.716	0.016480	11.25	2.61	0.33	0.00	27.25	31.70	1.87
1RHR	7,400	180.0	14.70	7.511	0.016510	11.25	2.61	0.33	0.00	25.40	31.70	2.65
1RHR	7,400	185.0	14.70	8.384	0.016540	11.25	2.61	0.33	0.00	23.35	31.70	3.51
1RHR	7,400	190.0	14.70	9.340	0.016572	11.25	2.61	0.33	0.00	21.10	31.70	4.44
1RHR	7,400	195.0	14.70	10.385	0.016604	11.25	2.61	0.33	0.00	18.63	31.70	5.47
1RHR	7,400	200.0	14.70	11.526	0.016637	11.25	2.61	0.33	0.00	15.91	31.70	6.59
1RHR	7,400	205.0	14.70	12.770	0.016670	11.25	2.61	0.33	0.00	12.94	31.70	7.81
1RHR	7,000	155.0	14.70	4.204	0.016368	11.25	2.34	0.30	0.00	33.35	29.50	-1.63
1RHR	7,000	160.0	14.70	4.741	0.016395	11.25	2.34	0.30	0.00	32.12	29.50	-1.11
1RHR	7,000	165.0	14.70	5.336	0.016422	11.25	2.34	0.30	0.00	30.76	29.50	-0.53
1RHR	7,000	170.0	14.70	5.993	0.016451	11.25	2.34	0.30	0.00	29.24	29.50	0.11
1RHR	7,000	175.0	14.70	6.716	0.016480	11.25	2.34	0.30	0.00	27.56	29.50	0.82
1RHR	7,000	180.0	14.70	7.511	0.016510	11.25	2.34	0.30	0.00	25.70	29.50	1.60
1RHR	7,000	185.0	14.70	8.384	0.016540	11.25	2.34	0.30	0.00	23.66	29.50	2.45
1RHR	7,000	190.0	14.70	9.340	0.016572	11.25	2.34	0.30	0.00	21.40	29.50	3.39
1RHR	7,000	195.0	14.70	10.385	0.016604	11.25	2.34	0.30	0.00	18.93	29.50	4.42
1RHR	7,000	200.0	14.70	11.526	0.016637	11.25	2.34	0.30	0.00	16.22	29.50	5.54
1RHR	7,000	205.0	14.70	12.770	0.016670	11.25	2.34	0.30	0.00	13.25	29.50	6.77

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature **2 RHR @ Max NPSHr** (>7 hrs of operation)

Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
2RHR	14,200	155.0	14.70	4.204	0.016368	11.25	3.95	1.22	0.00	30.82	30.00	-0.35
2RHR	14,200	160.0	14.70	4.741	0.016395	11.25	3.95	1.22	0.00	29.59	30.00	0.17
2RHR	14,200	165.0	14.70	5.336	0.016422	11.25	3.95	1.22	0.00	28.22	30.00	0.75
2RHR	14,200	170.0	14.70	5.993	0.016451	11.25	3.95	1.22	0.00	26.70	30.00	1.39
2RHR	14,200	175.0	14.70	6.716	0.016480	11.25	3.95	1.22	0.00	25.02	30.00	2.10
2RHR	14,200	180.0	14.70	7.511	0.016510	11.25	3.95	1.22	0.00	23.17	30.00	2.87
2RHR	14,200	185.0	14.70	8.384	0.016540	11.25	3.95	1.22	0.00	21.12	30.00	3.73
2RHR	14,200	190.0	14.70	9.340	0.016572	11.25	3.95	1.22	0.00	18.87	30.00	4.66
2RHR	14,200	195.0	14.70	10.385	0.016604	11.25	3.95	1.22	0.00	16.39	30.00	5.69
2RHR	14,200	200.0	14.70	11.526	0.016637	11.25	3.95	1.22	0.00	13.68	30.00	6.81
2RHR	14,200	205.0	14.70	12.770	0.016670	11.25	3.95	1.22	0.00	10.71	30.00	8.04
2RHR	12,800	155.0	14.70	4.204	0.016368	11.25	3.21	0.99	0.00	31.79	28.50	-1.39
2RHR	12,800	160.0	14.70	4.741	0.016395	11.25	3.21	0.99	0.00	30.56	28.50	-0.87
2RHR	12,800	165.0	14.70	5.336	0.016422	11.25	3.21	0.99	0.00	29.19	28.50	-0.29
2RHR	12,800	170.0	14.70	5.993	0.016451	11.25	3.21	0.99	0.00	27.68	28.50	0.35
2RHR	12,800	175.0	14.70	6.716	0.016480	11.25	3.21	0.99	0.00	26.00	28.50	1.06
2RHR	12,800	180.0	14.70	7.511	0.016510	11.25	3.21	0.99	0.00	24.14	28.50	1.83
2RHR	12,800	185.0	14.70	8.384	0.016540	11.25	3.21	0.99	0.00	22.09	28.50	2.69
2RHR	12,800	190.0	14.70	9.340	0.016572	11.25	3.21	0.99	0.00	19.84	28.50	3.63
2RHR	12,800	195.0	14.70	10.385	0.016604	11.25	3.21	0.99	0.00	17.37	28.50	4.66
2RHR	12,800	200.0	14.70	11.526	0.016637	11.25	3.21	0.99	0.00	14.65	28.50	5.78
2RHR	12,800	205.0	14.70	12.770	0.016670	11.25	3.21	0.99	0.00	11.68	28.50	7.01

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile -- Overpressure Required vs Pool Temperature **CS Max NPSHr** **(>7 hrs of operation)**

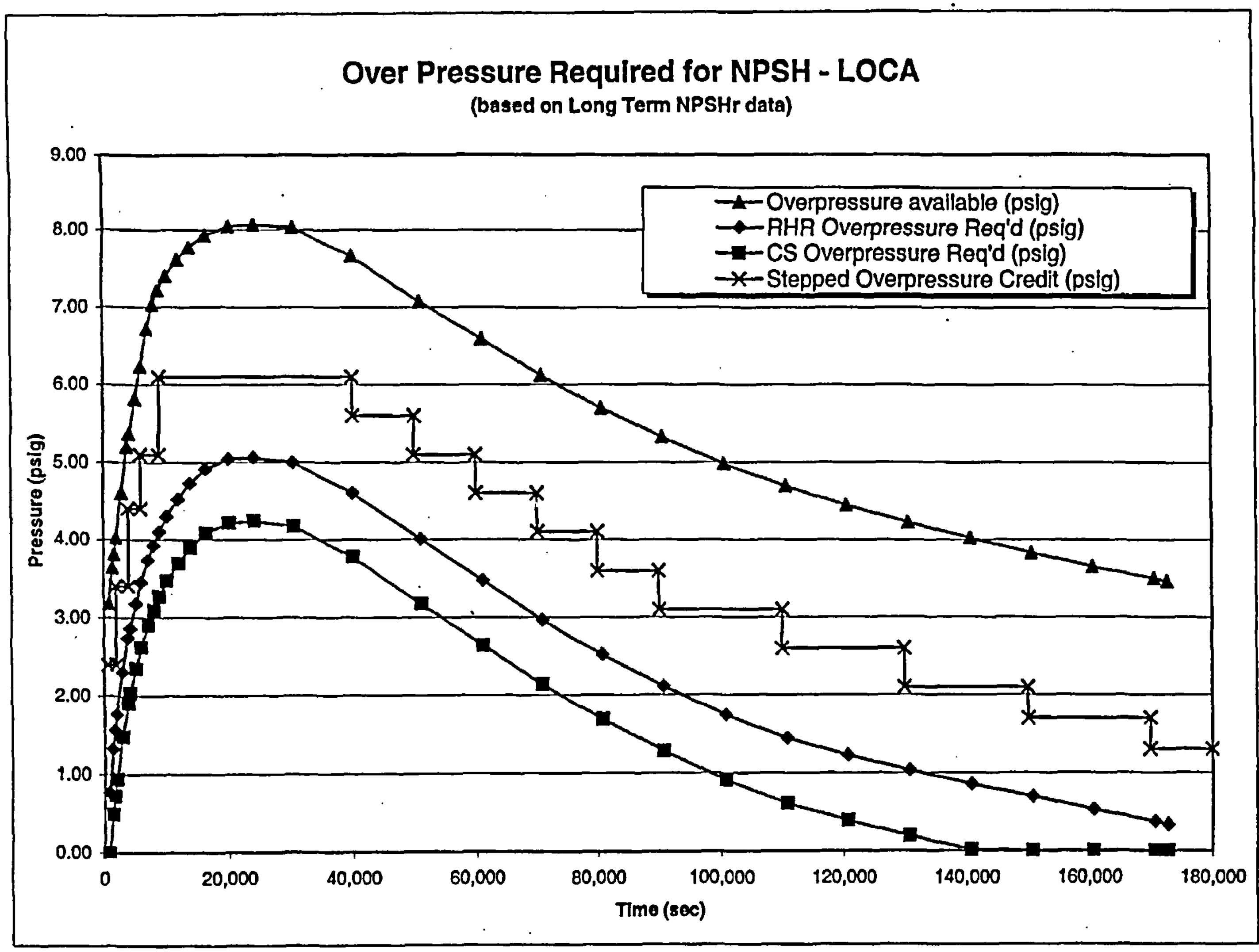
Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
CS	4,600	125.0	14.70	1.942	0.016225	11.42	5.29	0.51	0.00	35.43	35.00	-0.18
CS	4,600	130.0	14.70	2.223	0.016246	11.42	5.29	0.51	0.00	34.81	35.00	0.08
CS	4,600	135.0	14.70	2.537	0.016269	11.42	5.29	0.51	0.00	34.11	35.00	0.38
CS	4,600	140.0	14.70	2.889	0.016293	11.42	5.29	0.51	0.00	33.33	35.00	0.71
CS	4,600	145.0	14.70	3.282	0.016317	11.42	5.29	0.51	0.00	32.45	35.00	1.09
CS	4,600	150.0	14.70	3.718	0.016342	11.42	5.29	0.51	0.00	31.46	35.00	1.50
CS	4,600	155.0	14.70	4.204	0.016368	11.42	5.29	0.51	0.00	30.36	35.00	1.97
CS	4,600	160.0	14.70	4.741	0.016395	11.42	5.29	0.51	0.00	29.13	35.00	2.49
CS	4,600	165.0	14.70	5.336	0.016422	11.42	5.29	0.51	0.00	27.76	35.00	3.06
CS	4,600	170.0	14.70	5.993	0.016451	11.42	5.29	0.51	0.00	26.25	35.00	3.70
CS	4,600	175.0	14.70	6.716	0.016480	11.42	5.29	0.51	0.00	24.57	35.00	4.40
CS	4,600	180.0	14.70	7.511	0.016510	11.42	5.29	0.51	0.00	22.71	35.00	5.17
CS	4,600	185.0	14.70	8.384	0.016540	11.42	5.29	0.51	0.00	20.66	35.00	6.02
CS	4,600	190.0	14.70	9.340	0.016572	11.42	5.29	0.51	0.00	18.41	35.00	6.95
CS	4,600	195.0	14.70	10.385	0.016604	11.42	5.29	0.51	0.00	15.94	35.00	7.97
CS	4,600	200.0	14.70	11.526	0.016637	11.42	5.29	0.51	0.00	13.22	35.00	9.09
CS	4,600	205.0	14.70	12.770	0.016670	11.42	5.29	0.51	0.00	10.25	35.00	10.31

Table 4.4
General Profile - Overpressure vs Pool Temperature

General Profile – Overpressure Required vs Pool Temperature **CS Max NPSHr** **(>7 hrs of operation)**

Pump(s)	Flow, Q (gpm)	Pool Temp (F)	Pool Pressure psia	Pg (psia)	Vf (ft^3/lb)	Z (ft)	hf (ft)	hs (ft)	hd (ft)	NPSHa (ft)	NPSHr (ft)	OPR (psig)
CS	3,500	125.0	14.70	1.942	0.016225	11.42	3.06	0.29	0.00	37.88	29.60	-3.54
CS	3,500	130.0	14.70	2.223	0.016246	11.42	3.06	0.29	0.00	37.26	29.60	-3.27
CS	3,500	135.0	14.70	2.537	0.016269	11.42	3.06	0.29	0.00	36.56	29.60	-2.97
CS	3,500	140.0	14.70	2.889	0.016293	11.42	3.06	0.29	0.00	35.78	29.60	-2.63
CS	3,500	145.0	14.70	3.282	0.016317	11.42	3.06	0.29	0.00	34.90	29.60	-2.25
CS	3,500	150.0	14.70	3.718	0.016342	11.42	3.06	0.29	0.00	33.91	29.60	-1.83
CS	3,500	155.0	14.70	4.204	0.016368	11.42	3.06	0.29	0.00	32.81	29.60	-1.36
CS	3,500	160.0	14.70	4.741	0.016395	11.42	3.06	0.29	0.00	31.58	29.60	-0.84
CS	3,500	165.0	14.70	5.336	0.016422	11.42	3.06	0.29	0.00	30.21	29.60	-0.26
CS	3,500	170.0	14.70	5.993	0.016451	11.42	3.06	0.29	0.00	28.69	29.60	0.38
CS	3,500	175.0	14.70	6.716	0.016480	11.42	3.06	0.29	0.00	27.01	29.60	1.09
CS	3,500	180.0	14.70	7.511	0.016510	11.42	3.06	0.29	0.00	25.16	29.60	1.87
CS	3,500	185.0	14.70	8.384	0.016540	11.42	3.06	0.29	0.00	23.11	29.60	2.72
CS	3,500	190.0	14.70	9.340	0.016572	11.42	3.06	0.29	0.00	20.86	29.60	3.66
CS	3,500	195.0	14.70	10.385	0.016604	11.42	3.06	0.29	0.00	18.38	29.60	4.69
CS	3,500	200.0	14.70	11.526	0.016637	11.42	3.06	0.29	0.00	15.67	29.60	5.81
CS	3,500	205.0	14.70	12.770	0.016670	11.42	3.06	0.29	0.00	12.70	29.60	7.04

Figure 4.2



OPC overpressure credit	
(sec)	(psig)
601	2.4
2000	2.4
2001	3.4
4000	3.4
4001	4.4
6000	4.4
6001	5.1
9000	5.1
9001	6.1
40000	6.1
40001	5.6
50000	5.6
50001	5.1
60000	5.1
60001	4.6
70000	4.6
70001	4.1
80000	4.1
80001	3.6
90000	3.6
90001	3.1
110000	3.1
110001	2.6
130000	2.6
130001	2.1
150000	2.1
150001	1.7
170000	1.7
170001	1.3
180000	1.3

Figure 4.3

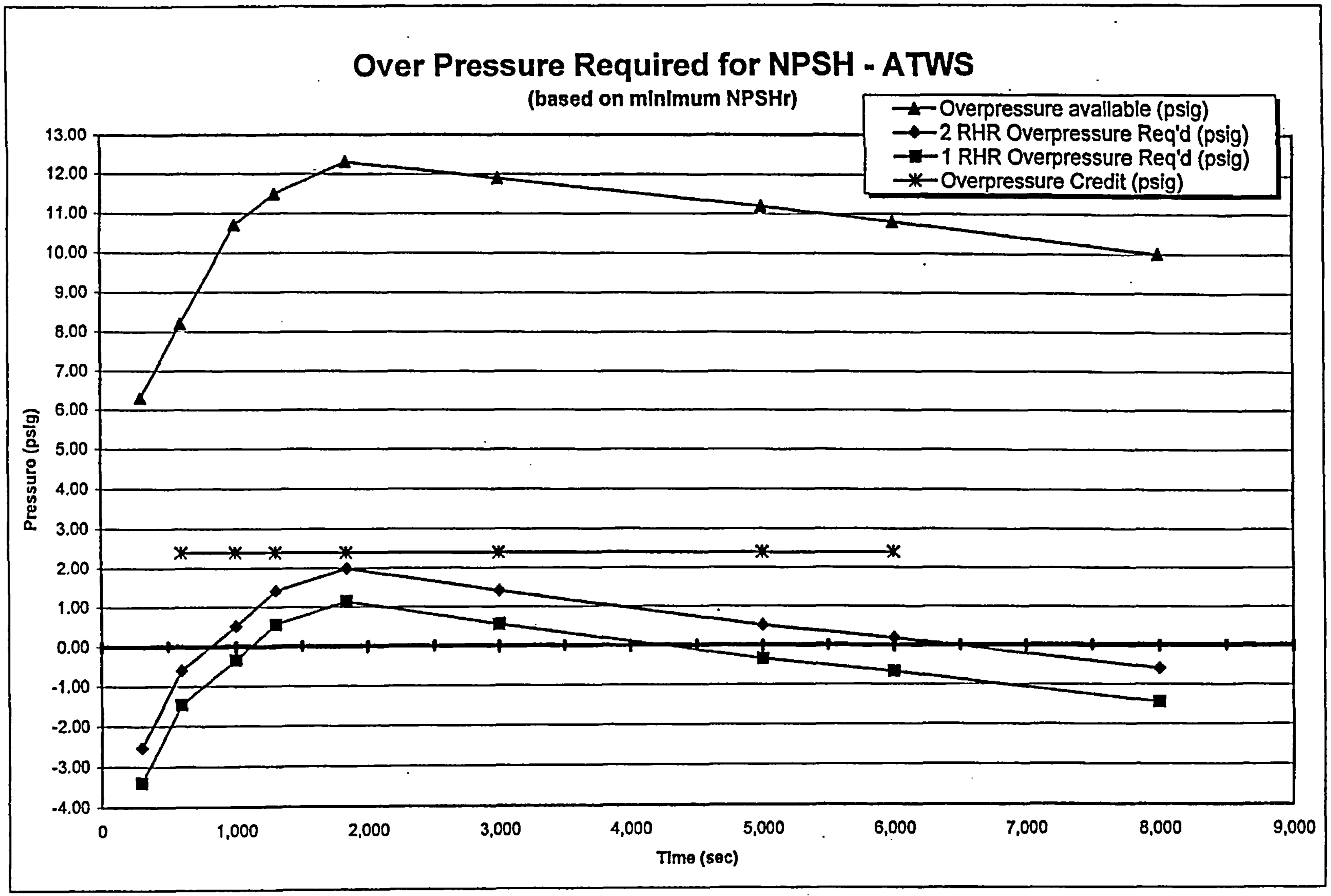

Over Pressure Required for NPSH - ATWS
(based on minimum NPSHr)
Overpressure available (psig)
2 RHR Overpressure Req'd (psig)
1 RHR Overpressure Req'd (psig)
Overpressure Credit (psig)
Pressure (psig)
Time (sec)
13.00
12.00
11.00
10.00
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
-3.00
-4.00
0
1,000
2,000
3,000
4,000
5,000
6,000
7,000
8,000
9,000

Figure 4.4-1

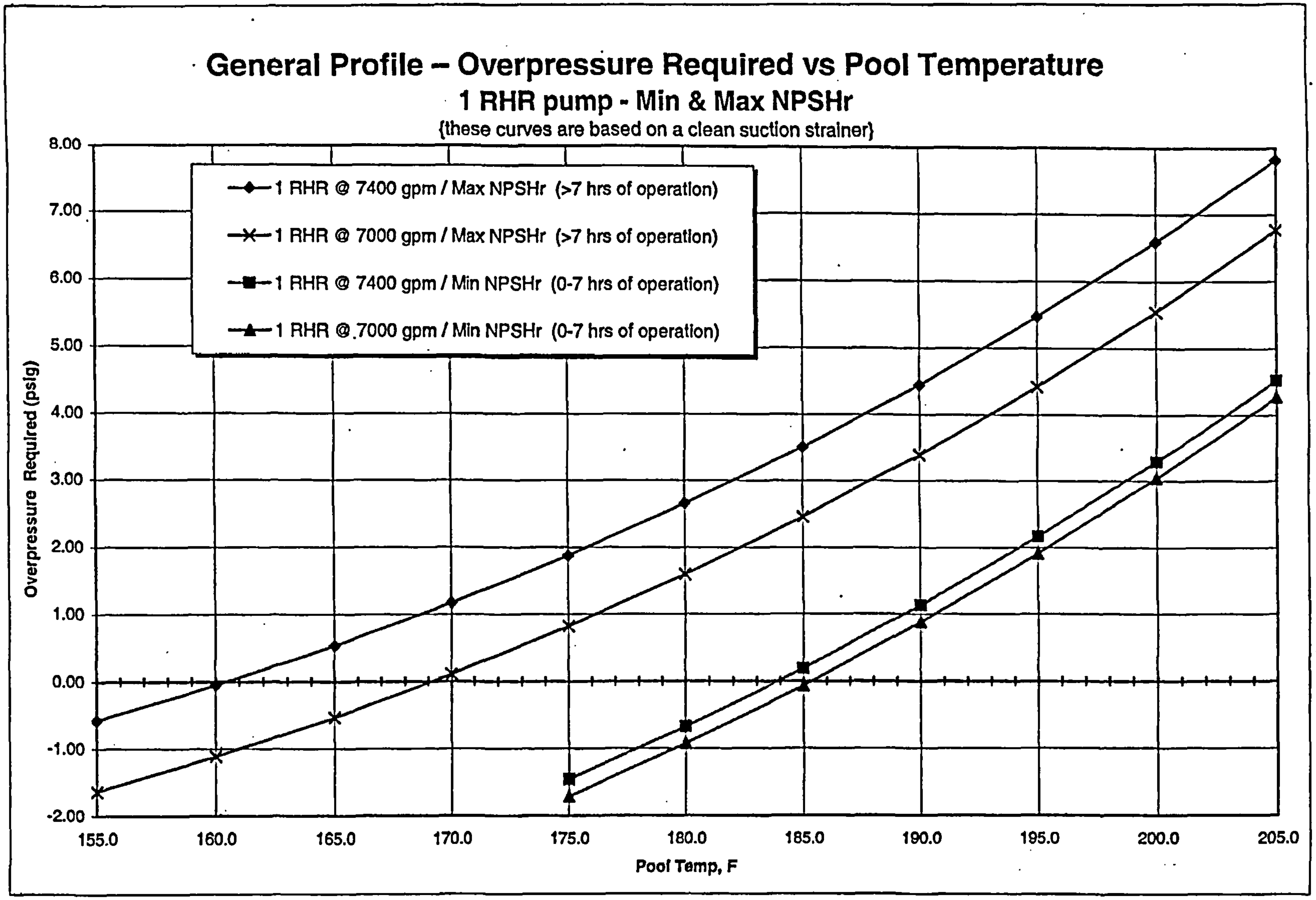
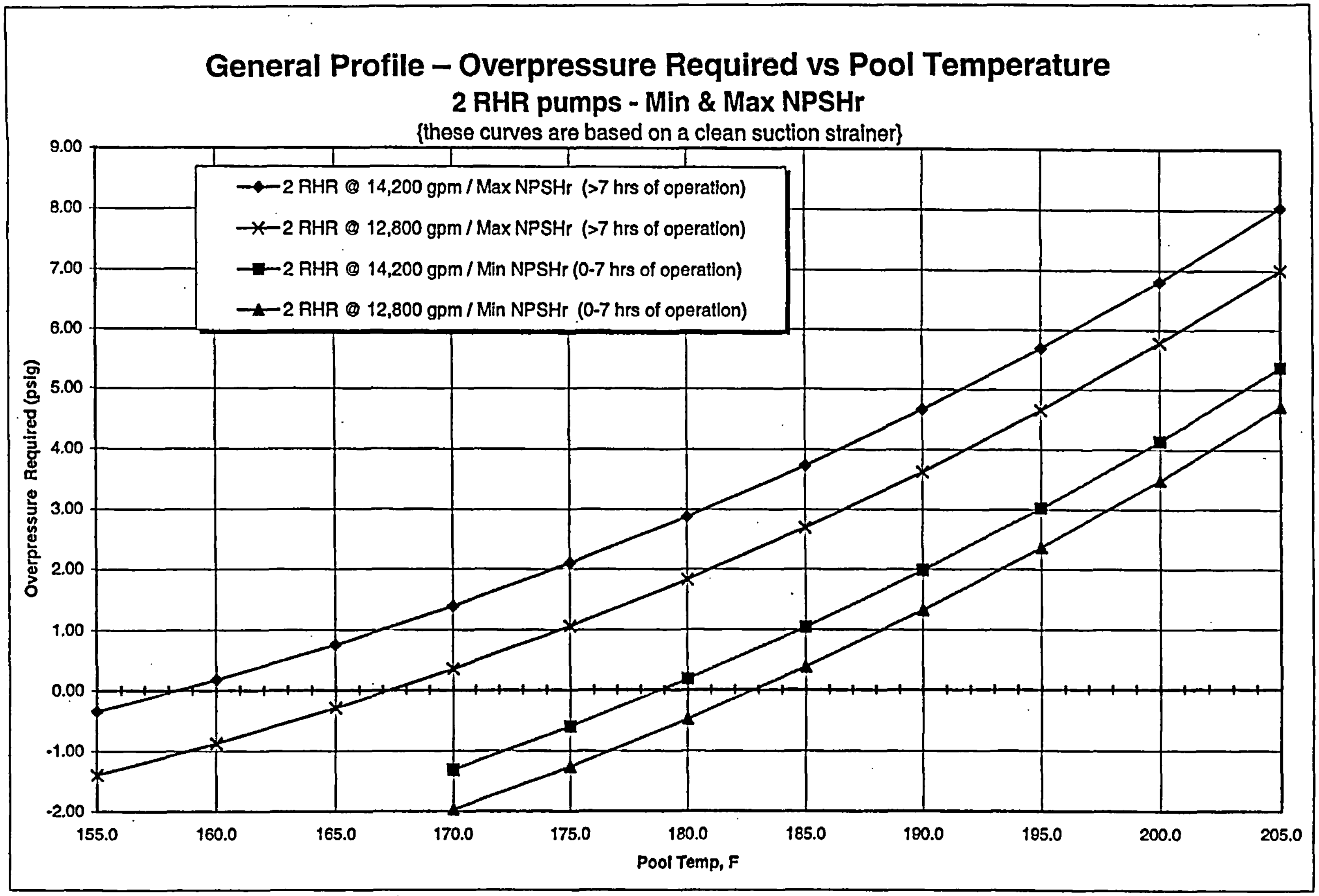

General Profile – Overpressure Required vs Pool Temperature
1 RHR pump - Min & Max NPSHr
{these curves are based on a clean suction strainer}
1 RHR @ 7400 gpm / Max NPSHr (>7 hrs of operation)
1 RHR @ 7000 gpm / Max NPSHr (>7 hrs of operation)
1 RHR @ 7400 gpm / Min NPSHr (0-7 hrs of operation)
1 RHR @ 7000 gpm / Min NPSHr (0-7 hrs of operation)
Overpressure Required (psig)
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
155.0
160.0
165.0
170.0
175.0
180.0
185.0
190.0
195.0
200.0
205.0
Pool Temp, F

Figure 4.4-2


General Profile – Overpressure Required vs Pool Temperature
2 RHR pumps - Min & Max NPSHr
{these curves are based on a clean suction strainer}
2 RHR @ 14,200 gpm / Max NPSHr (>7 hrs of operation)
2 RHR @ 12,800 gpm / Max NPSHr (>7 hrs of operation)
2 RHR @ 14,200 gpm / Min NPSHr (0-7 hrs of operation)
2 RHR @ 12,800 gpm / Min NPSHr (0-7 hrs of operation)
Overpressure Required (psig)
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
155.0
160.0
165.0
170.0
175.0
180.0
185.0
190.0
195.0
200.0
205.0
Pool Temp, F

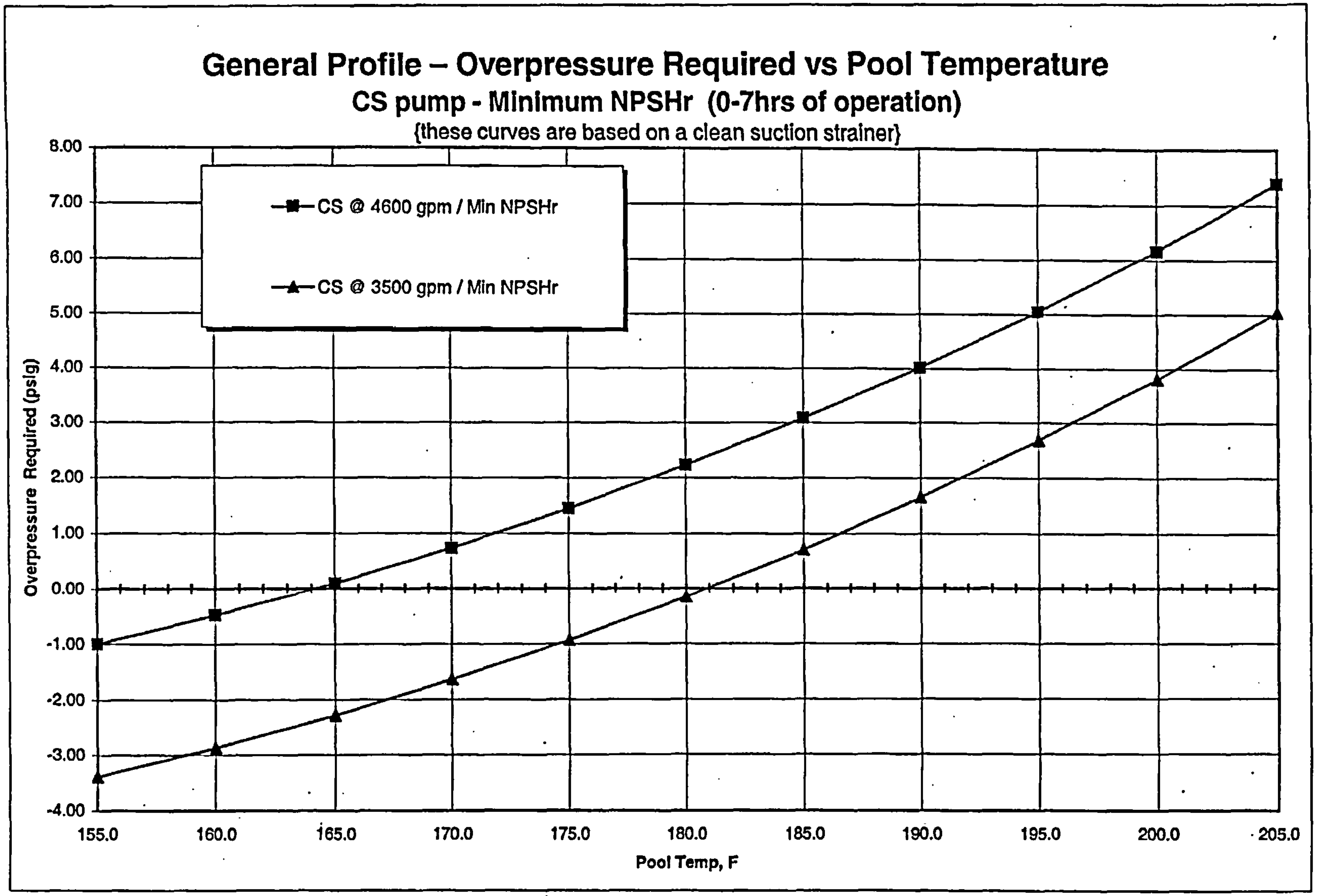
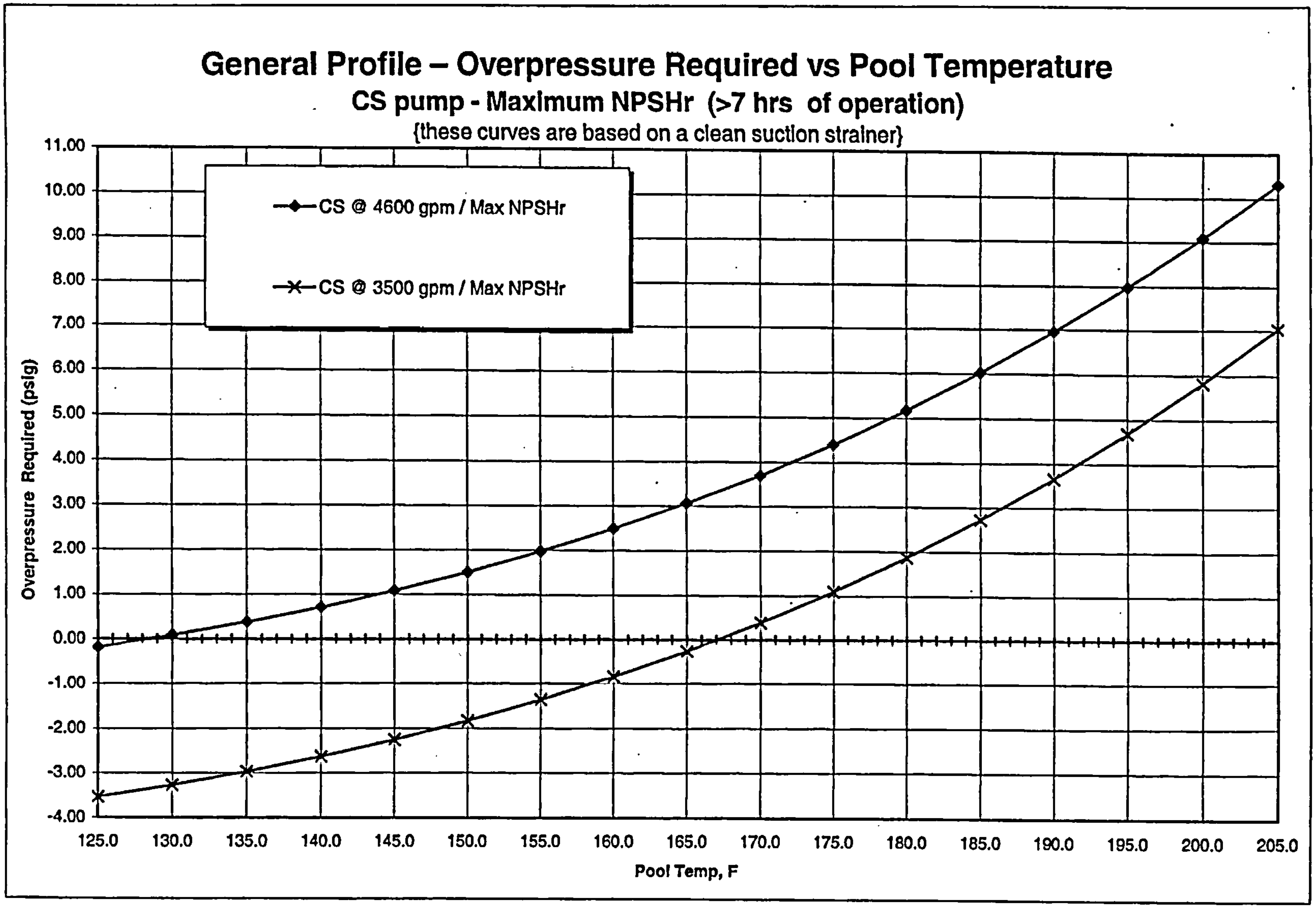

General Profile – Overpressure Required vs Pool Temperature
CS pump - Minimum NPSHr (0-7hrs of operation)
{these curves are based on a clean suction strainer}
CS @ 4600 gpm / Min NPSHr
CS @ 3500 gpm / Min NPSHr
Overpressure Required (psig)
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
-3.00
-4.00
155.0
160.0
165.0
170.0
175.0
180.0
185.0
190.0
195.0
200.0
205.0
Pool Temp, F

Figure 4.4-4


General Profile – Overpressure Required vs Pool Temperature
CS pump - Maximum NPSHr (>7 hrs of operation)
{these curves are based on a clean suction strainer}
CS @ 4600 gpm / Max NPSHr
CS @ 3500 gpm / Max NPSHr
Overpressure Required (psig)
11.00
10.00
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
-3.00
-4.00
125.0
130.0
135.0
140.0
145.0
150.0
155.0
160.0
165.0
170.0
175.0
180.0
185.0
190.0
195.0
200.0
205.0
Pool Temp, F

VY CALCULATION REVIEW FORM

Calculation Number: VYC-0808 Revision Number: 6 CCN Number: 4

Title: Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss of Coolant Accident **or Anticipated Transients Without Scram**

Reviewer Assigned: Edward Lind Required Date 8-12-03 .

☐ Interdiscipline Review ☒ Independent Review

Comments*	**Resolution**
Section 2 –Identify that previous revisions and CCNs of Calculation VYC-0808 had not not taken credit for torus air space pressure.	Section 2: 2nd paragraph revised as requested.
Section 4- Revise text referencing Figure 2.1-1 and 2.1.2 of Attachment 3 to calculation VYC-0808 to state that figures provide a plot of *Allowable Operating Periods @ NPSHa Specified* values for various flow rates not *NPSHr*	Section 4.0, 4.1, 4.2, and 4.3 revised as requested.
Calculation Title – Since this calc now addresses ATWS, in addition to LOCA, revise title to reflect ATWS.	Title on cover page and Attachment A revised to add ATWS.

Edward Lind *Edward Lind* /	8-12-03	Frank Capuano *F. Capuano* /	8-12-03
Reviewer Signature	Date	Calculation Preparer (Comments Resolved)	Date
Method of Review: ☒ Calculation/Analysis Review ☐ Alternative Calculation ☐ Qualification Testing		Edward Lind *Edward Lind* /	8-12-03
		Reviewer Signature (Comments Resolved)	Date

*Comments shall be specific, not general. Do not list questions or suggestions unless suggesting wording to ensure the correct interpretation of issues. Questions should be asked of the preparer directly.

DOCUMENTATION OF COMPUTER RESOURCE USE

CALCULATION NO.: VYC-0808 REVISION NO.: 6 CCN No.: 4

Computer Used (include manufacturer, CPU Type, and operating system version and level):

Not Applicable - See Below

Computer Input Attached*? ☐ Yes ☐ No

Location/Identifier:

Computer Output Attached*? ☐ Yes ☐ No

Location/Identifier:

* Large volume input/output should be provided on CD. See Appendix E for format requirements.

List the computer codes used, and complete the following:

	Approved per PP 7800		Appropriateness Verified		Outstanding SPRs or Code Errors[1]	
Code Name/Version and/or Script File	Yes[3]	No	Yes	No	Yes[2]	No
NONE **Only Microsoft Office software used, Word 97 SR-2 and Excel 97 SR-2**						

[1] Software Problem Report (SPR), does not exist as a reporting method in PP 7800 and AP 6030. Contact the Code sponsor and review any outstanding SPRs or Code errors. [ER2000805]
[2] If yes, fill out information below.
[3] If yes, include the Code name on the Computer Code line of the title page, VYAPF 0017.01.

If a computer code was not verified in accordance with PP 7800 and AP 6030, or if there are outstanding SPRs, state below why it is appropriate.

Code Name/Script File	Appropriateness

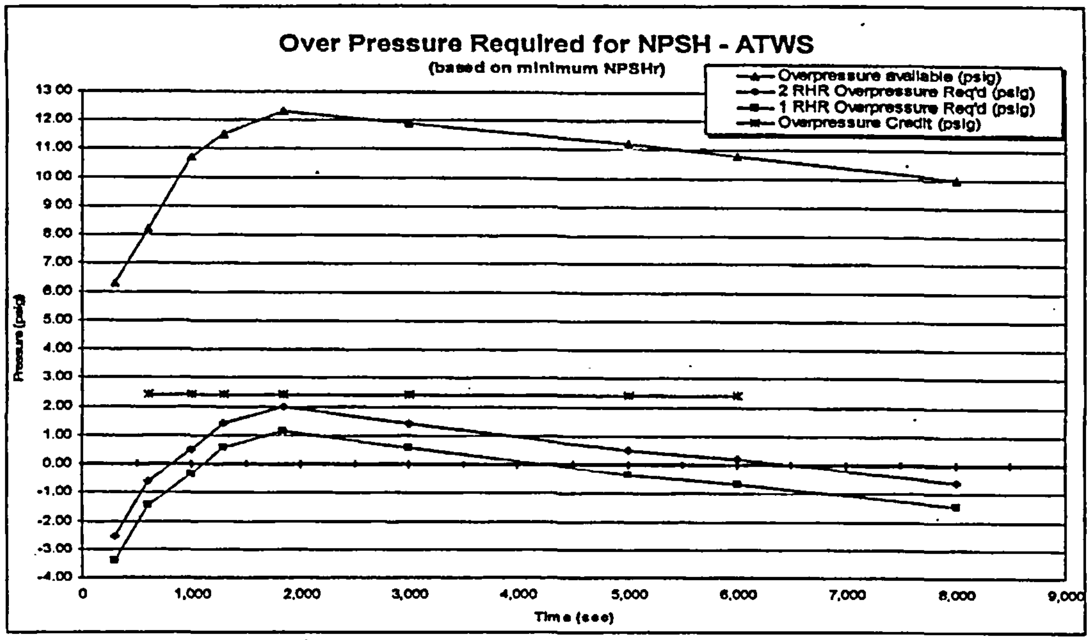
Comment No	Task Number	Reviewer	Comment	Resolution	Disposition
1843	VYC-0808	Paul Rainey	CCN cover sheet, Conclusion, Add word "required" after words "The overpressure credit" after 2nd bullet.	"required" incorporated on Cover sheet and in Section 6.0	Resolution accepted. B. Slifer 8/12/03
1844	VYC-0808	Paul Rainey	Section 2, p 8, typo next to last paragraph after Section.	Typo corrected: text revised to read *Section 2.4 of Calc ...* vs *Section .4 of Calc ....*	Resolution accepted. B. Slifer 8/12/03
1845	VYC-0808	Paul Rainey	Section 4, What is bases for determining OPC? Why isn't margin OPA-OPR?	Section 4.0: Added discussion about selection of OPC, excerpt follows: *OPC (psig)– Overpressure Credit Taken* *The overpressure credited in the evaluation of NPSH. Engineering judgement is used to select the credit to be greater than the OPR, by a reasonable amount, and less than the OPA.* While OPC-OPR is a simpler form of defining margin (NPSHcm), all discussion of NPSHcm is deleted throughout calc, as agreed on 8/7/03 tel-con discussion.	Resolution accepted. B. Slifer 8/12/03
1846	VYC-0808	Paul Rainey	Section 4.1, 2nd paragraph, discussion would lead you to believe that overpressure is required in short term. Add statement that overpressure is not required in short term.	Short Term Evaluation Section 4.1.1 concludes that overpressure is not required. The conclusion on the Cover sheet and Section 6.0 for Short Term LOCA is revised to state that overpressure is not required for the short term.	Resolution accepted. B. Slifer 8/12/03
1847	VYC-0808	Paul Rainey	Section 4.3, governing analysis should be based on 1 RHR pump in each loop. Plant doesn't run 2 RHR pumps in 1 loop while in torus cooling. NPSH required should be 23.8 ft since analysis is only run for 8000 seconds. With 1 pump per loop OPR is less than 2.4 psig.	As agreed to by tel-con on 8/7/03, analysis of two RHR pumps is maintained in the calculation for conservatism. However, Figure 4.3 is revised to add the 1 RHR pump OPR. This then clearly shows that there is additional NPSH margin for one pump operation. See excerpt of Fig 4.3 below. Since selection of OPC is somewhat arbitrary, see response to comment 1845, use of an OPC of 2.4 is appropriate. It is consistent with the limit used for LOCA (up to 2000 seconds) while enveloping for both 1 RHR and 2RHR pump operation.	Resolution accepted. B. Slifer 8/12/03

1848	VYC-0808	Paul Rainey	Section 5.3 and 5.4, revise per comment 1847.	Section 5.3 (<7hrs) revised to address NPSHa, OPR, OPC, and OPA for both 1 RHR and 2 RHR pumps. See excerpt of Section 5.3 below. Section 5.4 (>7hrs) revised to address both 1RHR and 2RHR pump operation, citing NPSHr at pertinent time increments and discussing the impact of the substantial rate of decrease of the pool temperature. See excerpt of Section 5.4 below.	Resolution accepted. B. Slifer 8/12/03
1849	VYC-0808	Paul Rainey	Section 5.5, add words "without strainer debris loading" to end of next to last sentence.	Incorporated words into Section 5.5 and Section 2.0. Excerpt follows: *The profiles are intended to serve as the basis for determining overpressure requirements when performing RHR and CS pump NPSH evaluation for any other events, which cause elevated suppression pool temperatures,* ***without strainer debris loading.***	Resolution accepted. B. Slifer 8/12/03
1850	VYC-0808	Paul Rainey	General comment, unless you're familiar with calc details the impression is left that you need to cut back flow at 10 minutes which is not the case. It would be helpful to state when we actually need to cut back flow for the various cases.	Section 4.2 revised to address flow rates evaluated. Section 4.2.1 revised to discuss throttling and reference generic profiles from Section 4.4. See excerpts of Section 4.2 and 4.2.1 below.	Resolution accepted. B. Slifer 8/12/03
1851	VYC-0808	Paul Rainey	Table 4.3 , see comment 1847.	1 RHR pump is added to Table 4.3 "NPSHa vs decreasing pool temperature". Refer to response to comments 1847 and 1848	Resolution accepted. B. Slifer 8/12/03
1852	VYC-0808	Paul Rainey	Fig. 4.2, per Bruce Slifer comment remove OPC graph and table.	As agreed to by tel-con on 8/7/03, OPC is maintained in Fig 4.2, however it is revised to be approximately midway between OPA and OPR, where practical. See excerpt of Fig 4.2 below.	Resolution accepted. B. Slifer 8/12/03
1853	VYC-0808	Paul Rainey	Fig. 4.3, base on 1 RHR pump per loop.	Refer to response to comment 1847.	Resolution accepted. B. Slifer 8/12/03

1857	VYC-0808	Bruce Slifer	Page 8, the debris head loss from CCN03 is used in lieu of the values from Rev. 6. CCN03 was an assessment of increased sludge. The calculated head loss was less than the head loss due to debris assumed in Rev. 6. The intent of CCN03 was not to replace the debris head loss in Rev. 6. This is considered margin that VY may want to reserve for input to maintenance decisions regarding torus cleaning intervals.	Revised to use conservative debris loading from Rev 6, in lieu of loading from CCN#3 CS long term (hd) @ 3500gpm changed to 0.21' RHR one pump (hd) changed to 0.33' Sections 2.0, 4.1, 4.2, Tables 4.1 and 4.2, and Fig 4.2 revised accordingly. See excerpt from Section 2.0 below.	Resolution accepted. B. Slifer 8/12/03
1858	VYC-0808	Bruce Slifer	Page 9, the draft ATWS report is listed as an open item. This task report is now final.	Final Task Report T0902 is referenced in lieu of Draft Task Report. All related Open items and Assumptions removed. Reference 5 updated, Assumption # 2 of Section 3.0 removed, Open Item List removed and TOC revised, Section 4.3 revised accordingly.	Resolution accepted. B. Slifer 8/12/03
1859	VYC-0808	Bruce Slifer	Page 10, the basis for determining Overpressure Credit Taken needs more discussion.	See response to comment 1845.	Resolution accepted. B. Slifer 8/12/03
1860	VYC-0808	Bruce Slifer	Page 11, as noted in comment 1857, the design values for debris losses should not be changed, i.e. use 0.33 ft for 1 RHR and 0.21 ft for CS at 3500 gpm.	See response to comment 1857.	Resolution accepted. B. Slifer 8/12/03
1861	VYC-0808	Bruce Slifer	Page 13, Section 4.2.1, as noted in item 3, need more discussion on basis for determining overpressure credit.	Section 4.2.1 is revised to refer to discussion in Section 4.0 for discussion of OPC. See response to comment 1845 above.	Resolution accepted. B. Slifer 8/12/03
1862	VYC-0808	Bruce Slifer	Page 13, Section 4.3, final task report for ATWS is available, revise section accordingly.	See response to comment 1858.	Resolution accepted. B. Slifer 8/12/03
1863	VYC-0808	Bruce Slifer	Page 15, Section 4.3.1, explain basis for choosing 2.4 psig for ovepressure credit.	Section 4.3.1 is revised to refer to discussion in Section 4.0 for discussion of OPC. See response to comment 1845 above. Basis for 2.4 psig credit is discussed in response to comment 1847 above.	Resolution accepted. B. Slifer 8/12/03

1864	VYC-0808	Bruce Slifer	Page 16, Section 4.3.2, compare the assumptions made about cooldown rates between 7 and 20 hours, and 20 and 100 hours, to the LOCA analysis results. The cooldown rates should be similar.	Section 4.3.2 revised to add LOCA comparison. See excerpt below.	Resolution accepted. B. Slifer 8/12/03
1865	VYC-0808	Bruce Slifer	Page 19, Section 5.0, show COP available, required, and credited in addition to NPSHr and NPSHcm. (NPSHcm is difficult to understand. Is this standard for plants crediting COP?)	As agreed to by tel-con on 8/07/03, NPSHcm is deleted throughout calc. NPSHa, OPA, OPR, OPC added to Sections 5.2 and 5.3. See excerpt of Section 5.2 below. Excerpt of Section 5.3 is provided below, in response to Comment 1848. Excerpt of Section 5.4 is provided below in response to Comment 1848. (OPA, OPR, OPC not applicable to this Section)	Resolution accepted. B. Slifer 8/12/03
1866	VYC-0808	Bruce Slifer	Page 20, Section 5.5, should not have 4 different Figure 4s. Either list sequentially, or use 4.4.1, 4.4.2, etc.	Figures are renumbered 4.4-1 thru 4.4-4	Resolution accepted. B. Slifer 8/12/03
1867	VYC-0808	Bruce Slifer	Page 20, Section 6.0, VYC-1628 is superceded by the GE analysis, therefore revise section accordingly.	As agreed to by tel-con on 8/7/03, existing notation that VYC-1628 may be superseded by GE EPU Analysis is adequate. No change required.	Resolution accepted. B. Slifer 8/12/03
1868	VYC-0808	Bruce Slifer	Page 20, provide source of reference to VYDC-2003-008.	As confirmed by email dated 811/03, YYDC-2003-008 is correct. No change required.	Resolution accepted. B. Slifer 8/12/03

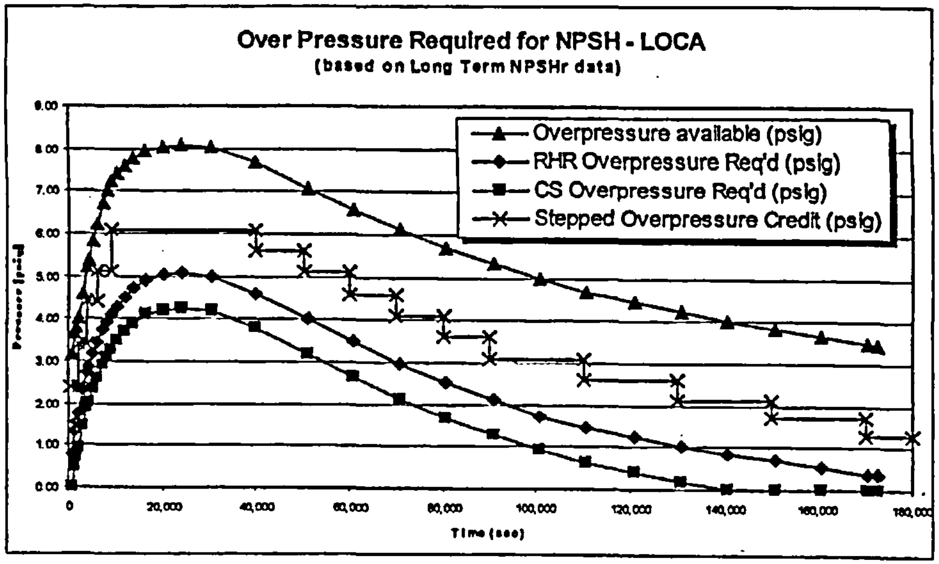
EXCERPTS FROM CCN #4 of VYC-0808 REFLECTING COMMENT INCORPORATION

<table>
<tr><th>Comment #</th><th>CCN Section</th><th>Excerpt</th></tr>
<tr><td>1847</td><td>Fig 4.3</td><td>
</td></tr>
<tr><td>1848</td><td>5.3</td><td>ATWS (<7hrs)
NPSHa is adequate for RHR pumps with an overpressure credit of 2.4 psig required between 600-6000 seconds, as shown in Fig. 4.3. NPSHa, OPR, OPC, OPA are shown below, at the peak temperature.

<table>
<tr><td>Pump</td><td>Total flow, gpm</td><td>NPSHr, ft</td><td>NPSHa, ft</td><td>OPR, psig</td><td>OPC, psig</td><td>OPA, psig</td></tr>
<tr><td>1RHR</td><td>7,400</td><td>23.8</td><td>21.1</td><td>1.1</td><td>2.4</td><td>12.3</td></tr>
<tr><td>2RHR</td><td>14,200</td><td>23.6</td><td>18.9</td><td>2.0</td><td>2.4</td><td>12.3</td></tr>
</table>

Note that for the same time period, this overpressure credit (2.4 psig) is bounded by that required for LOCA Long Term (2.4 to 4.4 psig).</td></tr>
</table>

1848	5.4	**ATWS (>7hrs)** NPSHa is adequate for the RHR pumps without crediting overpressure. This is due to the significant rate of reduction of the suppression pool temperature over time. The reduction in temperature, and associated vapor pressure, will more than offset increases in NPHr at extended operating times. Note that between 3000-8000 seconds (the end of tabulated data provided by GE) the pool temperature decreases to 175F, at the substantial uniform rate of about 2.4F/1000sec. The increased NPSHr, at significant time intervals, is shown below along with the approximate pool temp, in parentheses, at or below which, over pressure credit is not required. Pump / Total flow, gpm / 0-7 hr NPSHr, ft / 20 hr NPSHr, ft / >100 hr NPSHr, ft 1RHR / 7,400 / 23.8 / 30.5 (164F) / 31.7 (160F) 2RHR / 14,200 / 23.6 / 29.0 (162F) / 30.0 (158F)
1850	4.2	**LOCA - Long Term** The evaluation of NPSH is documented in Table 4.2 using a selected T/P points representing the long term profile of the suppression pool. The details of the evaluation are presented at the top of the Table followed by a matrix of the NPSH results for the T/P profile of CS and RHR. **The evaluated long term flow rates of 7400 gpm (RHR) and 3500 gpm (CS) are consistent with calculation VYC-0808 Rev 6 (Ref: 4).**
1850	4.2.1	**Evaluation** As can been seen from Figure 4.2 the overpressure required for RHR envelopes that required for CS and the overpressure varies continuously over time. In order to facilitate reporting and presentation of the overpressure required, an enveloping, stepped, overpressure credit is overlaid on Figure 4.2. **Refer to Section 4.0 for discussion on selection of overpressure credit.** **Though the long term flow rates are postulated at time 600 seconds (e.g. CS throttled down from 4600gpm to 3500gpm), it is not the intent of this calculation to imply at what time throttling should commence or how much throttling is required. This is a function of the time dependent NPSHr and pool temperature. This calculation conservatively evaluates the maximum NPSHr as occurring over the entire operating period (>600 sec). The actual NPSHr is lower between 0-7 hrs and increases after 7 hrs.** **Note that Section 4.4 develops required overpressure for both the CS and RHR pumps as a function of flow, temperature and NPSHr without any debris loading. Refer to Table 4.4 and Figures 4.4-1 to 4.4-4.**

1852	Fig 4.2	
1857	2.0	**<u>Pump Suction Strainer Head Loss during a LOCA</u>** As documented in ERC No. 2003-027 (Ref: 8), EPU does not affect the debris source terms developed in VYC-1677. Therefore the limiting head loss due to debris loading on the RHR and CS suction strainers remains the same as addressed in VYC-0808 Rev 6 including CCN #3 (Ref: 3). CCN #3 dispositions the up-to-date suction strainer head loss calculated in CCN #2 of VYC-1924 Rev 0 (Ref: 10) which in turn is based on the up-to-date debris source term information per VYC-1677 Rev 0 CCN #3. **Note that the debris loading from calc VYC-0808 Rev 6, as extracted from calc VYC-1924 Rev 0 (Ref: 2), is slightly larger than that documented in CCN #3. For conservatism, the larger debris loading is used.**

1864	4.3.2	Intermediate NPSHr Between 7-20 hrs, the NPSHr for 2 RHR pump operation increases from 23.6' to 29.0'. Conservatively ignoring this ramp up, the NPSHr at 7 hrs (25,200 seconds) is taken as 29.0'. As can be inferred from Table 4.3 "NPSHa vs. decreasing pool temperature" the pool only needs to cool about 13F to 162F over 17,200 seconds, to achieve an adequate NPSHa of about 29.0' without crediting any overpressure. This equates to a cooldown rate of about 0.8F/1000 seconds, **which is much less than the exhibited cooldown rate between 3000-8000 seconds.** **(Note that if the 20 hour (72,000 sec) time is used, then the required cooldown rate becomes 13F over 64,000 sec or about 0.2F/1000 sec.)** In consideration of the conservative selection of NPSHr (@ 7 hrs) and the minimal cooldown rate required, there is adequate NPSHa without crediting overpressure. Maximum NPSHr Between 20-100 hrs, the NPSHr for 2 RHR pump operation increases from 29.0' to 30.0'. Conservatively ignoring this ramp up, the NPSHr at 20 hrs (72,000 seconds) is taken as 30.0'. As can be inferred from Table 4.3 "NPSHa vs. decreasing pool temperature" the pool only needs to cool about 4F to 158F between 7hrs and 20 hrs (46,800 seconds), to achieve an adequate NPSHa of about 30.0' without crediting any overpressure. This equates to a cooldown rate of about 0.1F/1000 seconds. **Suppression Pool Cooldown Comparison to LOCA** **The temperature profile for the suppression pool during a LOCA is developed in GE-VYNPS-AEP-177 (Ref: 1). The pool temperature rises until about 24,000 seconds (~7 hrs). Shortly after peaking, there is essentially a uniform decrease in temperature between ~9hrs (194.1F) and 30hrs (174.4F) of about 0.2F/1000 sec. Between ~30hrs (174.4F) and the end of the tabulated data at ~48hrs (166.2F), the uniform decrease in temperature is about 0.1F/1000sec. The required suppression pool cooldown rates, projected above for ATWS, are similar to the suppression pool cooldown rates developed for LOCA.**
1865	5.2	**LOCA - Long Term (>600 sec):** NPSHa is adequate for both CS and RHR pumps with an overpressure credit that varies over time, as shown in Fig. 4.2. NPSHa, OPR, OPC, OPA are shown below, at the peak temperature Pump / Total flow, gpm / NPSHr, ft / NPSHa, ft / OPR, psig / OPC, psig / OPA, psig CS / 3,500 / 29.6 / 19.5 / 4.2 / 6.1 / 8.1 1RHR / 7,400 / 31.7 / 19.6 / 5.1 / 6.1 / 8.1

MINOR CALCULATION CHANGE FORM (Tracked via DRN in MERLIN)

Page 1 of 29

Calculation No.: VYC-2314	**Revision:** 0
Calculation Title: Minimum Containment Overpressure for Non-LOCA Events	**Indicate Status of Minor Calculation Change:** ~ Prel; Pend ~As-Built
MERLIN DRN No. or Minor Calculation Change No.: 01	
Modification No./Task No./ER No. VYDC-2003-008 (EPU Design Change).	
Computer Code Used ☐ Yes ☒ No. **If "Yes", Code:**	

1 Purpose of Change:	Address CR-VTY-2004-01974 and correct typographical errors
2 SSC affected:	None
3 Design Input Documents not used in parent Calculation:	See attached VYAPF0017.07
4 Drawings/Procedures/ Calculations / other Documents affected	See conclusion and VYAPF0017.07
5 Description of Change:	See attached description and replacement pages
6 Impact on existing calculation conclusion:	None
7 Impact on DBD's, UFSAR, Technical Specifications:	See Attachment 1 VYAPF0017.07 Form

8. The existing calculation does/does not (circle one) have a calculation verification checklist. See remark.

Remarks: This ENN-DC-126 MCC is to a VY design verified calculation prepared under AP-0017. AP-0017 did not have a "checklist", instead design verification was documented on form VYAP0017.04.

NOTE:

A. If UFSAR or Technical Specifications need to be revised, Minor Calculation Change Form should not be used unless it is an editorial change to the UFSAR or Technical Specifications.

B. Minor Calculation Change Forms do not change the status of the Parent Calculation Revision.

	Print/Sign	Date
Prepared by:	Pedro B. Perez [signature]	6/30/04
Reviewed by: *	Bruce C. Slifer [signature]	6/30/04
Approved by	James G. Rogers [signature]	6/30/04

* Where the original calculation was design verified, the reviewer signature confirms the latest design verification is still valid.

This IS a Quality Record -

Docket No. 50-271
DPS Exhibit #28
29 Pages

Section 5. Description of Change

Discussion

The minor change to VYC-2314 addresses condition report CR-VTY-2004-01974 [Reference 18]. The condition report documents a difference in primary containment leakage rates used in the Alternative Source Term (AST) and Extended Power Uprate (EPU) license amendment requests [References 19 and 20, respectively]. The EPU containment response analysis assumed a primary containment leakage rate of 0.8 wt-% per day corresponding to the current La in the Technical Specifications (Assumption 3 in VYC-2314). The AST proposed license amendment included an Appendix J exemption to remove the MSIV leakage term from the La term. The AST analysis supported maintaining the La of 0.8 wt-% per day and increasing the MSIV leakage to an aggregate of 124 scfh. In addition, the AST analysis assumed a secondary containment bypass leakage of 5 scfh as part of the La term. The overall primary containment leakage rate used in the EPU containment response and net positive suction head (NPSH) calculations should be one that bounds the AST value.

The total AST primary containment leakage is obtained by adding the MSIV and La components. The unit conversion of the MSIV mass flow rate to primary containment wt-%/day is straight forward recognizing the 124 scfh is already at the post-LOCA accident pressure of 44 psig (UFSAR Section 5.2.4.5 [Reference 23]).

$$124\,scfh \times \left[\frac{14.7\ psia}{(44+14.7)\,psia}\right] \times \left[\frac{(460+338)R}{(460+68)R}\right] = 46.9\ cfh$$

$$46.9\frac{ft^3}{h} \times \left[\frac{24\,h}{d}\right] \times \left[\frac{1}{232{,}302\ ft^3}\right] \times 100\% = 0.5\ wt-\%\ per\ day$$

The drywell temperature corresponds to the maximum from a small steam line break and bounds all accidents [Reference 7]. The primary containment volume of 232,302 ft^3 is obtained by adding the minimum drywell and suppression pool free volumes of Reference 21. Standard temperature and pressure used in the AST application are 68°F and 14.7 psia, respectively [Reference 19].

The total leakage is then 0.8+0.5 or 1.3 wt-% per day and a bounding value of 1.5 wt-% per day is used for NPSH calculations that credit containment overpressure.

The increased leakage does not adversely affect the primary containment pressure response used in the net positive suction head evaluations. GE performed an evaluation for the post-LOCA containment response with the increased primary containment leakage rate of 1.5 wt-% per day [Reference 22]. The results of this evaluation were compared to the 0.8 wt-% per day results [Reference 7]. The increased leakage rate results in approximately 0.2 psi decrease at about 48 hours. Table I provides a comparison of the suppression pool pressure results.

The conclusions from the GE post-LOCA results are applicable to the non-LOCA calculations in VYC-2314. Leakage rate or flow is proportional to the square root of the ΔP (conversely, $\Delta P \propto w^2$). The post-accident LOCA ΔP (drywell to atmosphere) bounds the non-LOCA cases. As a result, the 0.2 psi decrease at 48 hours post-LOCA is also bounding for the non-LOCA events. The "Summary of Results" in VYC-2314 Revision 0

documents a the margin available for the non-LOCA NPSH evaluations is in the order of 3 to 5 psig. A maximum decrease of 0.2 psi will still provide adequate NPSH margin at the 8 hour time of interest.

Table I
Suppression Pool Pressure Comparison

Time (Seconds)	Suppression Pool Pressure (psia)		ΔP (psi)	Comment
	0.8 wt-% / day	1.5 wt-% / day		
0	14.7	14.7	0.0	
138.14	30.22	30.22	0.0	Suppression pool peak pressure
8035.1	21.73	21.72	-0.01	
24093.8	22.77	22.74	-0.03	Suppression pool peak temperature (6.7 hr)
51025.32	21.77	21.71	-0.06	
100697.01	19.68	19.56	-0.12	
150684.83	18.53	18.36	-0.17	
171934.83	18.18	17.98	-0.20	48 hr

Assumption 3 in VYC-2314 is modified to reflect the 1.5 wt-% per day primary containment bounding leakage rate.

3. Containment leakage has an insignificant impact on torus pressure. Maximum leakage is not to exceed 1.5% per day (bounding value) under post-LOCA conditions when pressures are maximum. The time frame of interest in the present calculation is approximately 8 hours, therefore leakage would be much less than 1.5% and can be ignored in the calculations.

Other changes in this MCC address typographical errors and do not change the conclusions of the calculation.

VYC-2314 Revision 0, Page 10 or 11 includes the following:

> *The limiting torus temperature from Reference 11 is shown on Figure 3. It is plotted along with the bounding torus temperature from the DBA LOCA analysis [Ref. 7]. The DBA LOCA curve bounds the Appendix R curve.*

The reference to Figure 3 above should be to Figure 4 and this typographical error is corrected below:

> The limiting torus temperature from Reference 11 is shown on Figure 4. It is plotted along with the bounding torus temperature from the DBA LOCA analysis [Ref. 7]. The DBA LOCA curve bounds the Appendix R curve.

VYC-2314 Revision 0, Figure 3 final peak pool temperature at 26800 seconds was 177.7°F it should be 177.4°F at 24560 seconds. The reference for the pool temperature (Note 1) should be to Reference 11. This correction has no impact on the results or conclusions in the calculation.

These corrections are reflected below:

FIGURE 3 (table only)
Alternate Shutdown

Time	T_pool (Note 1)	Time	RHR, 7000 gpm NPSHR (Note 2)	NPSHA (Note 3)	Margin
sec	F	hr	ft	ft	ft
0	90	0.0	23.5		
5184	155	1.4	23.5		
8640	165	2.4	23.5	30.8	
11520	170	3.2	23.5	29.2	5.7
17280	175	4.8	23.5	27.5	4.0
26800	177.4	7.4	23.5	26.7	3.2
36000	175	10.0	24.5	27.5	3.0
37440	170	10.4	24.7	29.2	4.5

Note 1 Pool temperature is from VYC-2306 [11].
Note 2 Required NPSH is from page 18, Attachment 5, VYC-0808, Rev. 6 [13].
Note 3 Available NPSH is from page 30 of CCN04 to VYC-0808, Rev. 6 [1].

VYC-2314 Revision 0, Figure 4 final peak pool temperature at 20260 seconds was 190°F. The temperature and time should be 189.5°F at 20190 seconds. This correction has no impact on the results or conclusions in the calculation.

These corrections are reflected below:

FIGURE4 (table only)
Appendix R

Time	T_pool (Note 1)	Time	RHR, 7000 gpm NPSHR (Note 2)	NPSHA (Note 3)	Margin	COPR (Note 4)	Isolate 120 F COPA (Note 5)
sec	F	hr	ft	ft	ft	psi	psi
0	90	0	23.5				
3600	126	1.0	23.5				0.3
7200	155	2.0	23.5				2.3
10800	173	3.0	23.5	28.2	4.7		4.0
14400	180	4.0	23.5	25.7	2.2	0.0	4.7
18000	188	5.0	23.5	22.3	-1.2	0.5	5.7
20190	189.5	5.6	23.5	21.3	-2.2	0.9	5.9
25200	188	7.0	23.5	22.3	-1.2	0.5	5.7

Conclusion

This MCC does not change the results in the existing calculation (VYC-2314, Revision 0). The MCC documents the primary containment leakage rate assumption of 1.5 wt-% per day. The need for crediting suppression pool overpressure in the RHR and CS NPSH evaluation shall also be addressed in the SADBD (Reference 17), UFSAR (Reference 23), system DBDs RHR (Reference 24) and CS (Reference 25), NEDC-33090P (Reference 26) and VYDC-2003-008 (Reference 27). These documents are listed in the attached AP0017.07 form as design output documents.

Note that the changes to the UFSAR were originally proposed in VYDC-2003-008 and are pending incorporation via the design change and licensing processes. The use of overpressure credit must be approved by the NRC as part of EPU. This MCC simply updates a design input parameter in the calculation that was performed under AP-0017.

No specific 50.59 Screening/Evaluation is required for this CCN since all EPU design changes and associated 50.59 documentation will be part of VYDC-2003-008 (Reference 27).

References

The following references support this MCC. These references are also listed in the attached VYAPF0017.07 form (Attached) and supplement the existing calculation design input and ouputs.

	Design Input Documents
18	CR-VTY-2004-01974, EPU Containment Analysis Inconsistent with AST Assumption, 6/15/04.
19	BVY 03-70, Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262, Alternative Source Term, 7/31/03.
20	BVY 03-80, Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate, 9/10/03.
21	ERC-2003-008, Design Input Request (DIR) for T400 Containment System Response for EPU / MELLLA+ (OPL-4A, Revision 1), 2/26/03
22	GE Letter, GE-VYNPS-AEP-346 (DRF 0000-0007-5271), VYNPS EPU T0400 : DBA-LOCA for Long Term NPSH Evaluation, 6/10/04
23	VYNPS Updated Final Analysis Report, Revision 18

	Design Output Documents
17	Safety Analysis Design Basis Document, Revision 2
23	Updated Final Analysis Report, Revision 18
24	Residual Heat Removal Design Basis Document, Revision 2
25	Core Spray Design Basis Document, Revision 1
26	NEDC-33090P, VYNPS Extended Power Uprate
27	VYDC-2003-008. Extended Power Uprate Design Change

Attachments

1 VYAPF0017.07
2 Reference 5 Table and Figures

Attachment 1

VYAPF0017.07 Form

2 Pages

VYC-2314

VY CALCULATION DATABASE INPUT FORM

Place this form in the calculation package immediately following the Title page or CCN form.

VY Calculation/CCN Number	Revision Number	Vendor Calculation Number	Revision Number
VYC-2314	0	N/A	0

Vendor Name: N/A PO Number: N/A

Originating Department Extended Power Uprate

Critical References Impacted: ☒ UFSAR ☒ DBD ☐ Reload. "Check" the appropriate box if any critical document is identified in the tables below.

EMPAC Asset/Equipment ID Number(s): N/A

EMPAC Asset/System ID Number(s): N/A

Keywords: Suppression Pool Temperature, Net Positive Suction Head, NPSH, Alternate Cooling, Appendix R, Station Blackout, Containment Overpressure

For Revision/CCN only: Are deletions to General References, Design Input Documents or Design Output Documents required? ☐ Yes† ☒ No

Design Input Documents and General References - The following documents provide design input or supporting information to this calculation. (Refer to Appendix A, sections 3.2.7 and section 4)

* Reference #	** DOC #	REV #	***Document Title (including Date, if applicable)	Significant Difference Review ††	**** Affected Program	Critical Reference (✓)
18	CR-VTY-2004-01974	0	EPU Containment Analysis Inconsistent with AST Assumption, 6/15/04.			
19	BVY 03-70	0	Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 262, Alternative Source Term, 7/31/03			
20	BVY 03-80	0	Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate, 9/10/03.			
21	ERC-2003-008	0	Design Input Request (DIR) for T400 Containment System Response for EPU / MELLLA+ (OPL-4A, Revision 1), 2/26/03			
22	GE-VYNPS-AEP-346	0	GE Letter, (DRF 0000-0007-5271), VYNPS EPU T0400 : DBA-LOCA for Long Term NPSH Evaluation, 6/10/04			
23	UFSAR	18	Updated Final Safety Analysis Report			✓

VY CALCULATION DATABASE INPUT FORM

Design Output Documents - This calculation provides output to the following documents. (Refer to Appendix A, section 5)

* Reference #	** DOC #	REV #	Document Title (including Date, if applicable)	**** Affected Program	†††Critical Reference (✓)
17	SADBD	2	Safety Analysis Design Basis Document		✓
24	RHRDBD	2	Residual Heat Removal System Design Basis Document		✓
25	CSDBD	1	Core Spray System Design Basis Document		✓
23	UFSAR	18	UFSAR, Section 6.5		✓
26	NEDC-33090	0	NEDC-33090P VYNPS Extended Power Uprate Report		
27	VYDC-2003-08	0	Extended Power Uprate Design Change		

* Reference # - Assigned by preparer to identify the reference in the body of the calculation.

** Doc # - Identifying number on the document, if any (e.g., 5920-0264, G191172, VYC-1286)

*** Document Title - List the specific documentation in this column. "See attached list" is not acceptable. Design Input/Output Documents should identify the specific design input document used in the calculation or the specific document affected by the calculation and not simply reference the document (e.g., VYDC, MM) that the calculation was written to support.

**** Affected Program List the affected program or the program that reference is related to or part of.

† If "yes," attach a copy of "VY Calculation Data" marked-up to reflect deletion (See Section 4.3.1.8 for Revision and 4.5.2.4.16 for CCNs).

†† If the listed input is a calculation listed in the calculation database that is not a calculation of record (see definition), place a check mark in this space to indicate completion of the required significant difference review. (see Appendix A, section 4.1.4.4.3). Otherwise, enter "N/A."

††† If the reference is UFSAR, DBD or Reload (IASD or OPL), check Critical Reference column and check UFSAR, DBD or Reload, as appropriate, on this form (above).

Attachment 2

GE-VYNPS-AEP-346 Data
(Reference 5)

Long-term NPSH Tabular Data

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
0.00	14.70	90.0	6.800E+04
138.14	30.22	149.2	7.681E+04
236.79	24.87	157.4	7.683E+04
353.20	20.34	162.5	7.775E+04
500.01	18.65	166.1	7.823E+04
552.04	18.49	167.1	7.829E+04
778.85	17.88	169.6	7.921E+04
1088.64	18.12	171.7	7.926E+04
1400.32	18.34	173.6	7.931E+04
1711.20	18.51	175.2	7.933E+04
2022.07	18.72	176.6	7.936E+04
2331.73	18.90	177.8	7.937E+04
2641.79	19.11	178.9	7.939E+04
2950.64	19.30	180.0	7.940E+04
3259.07	19.52	180.9	7.941E+04
3567.10	19.72	181.8	7.940E+04
3875.54	19.89	182.6	7.940E+04
4185.20	20.07	183.3	7.941E+04
4496.29	20.21	184.0	7.943E+04
4805.95	20.37	184.6	7.943E+04
5108.82	20.50	185.1	7.944E+04
5411.17	20.63	185.7	7.944E+04
5704.57	20.78	186.1	7.944E+04
5996.57	20.93	186.6	7.944E+04
6241.48	21.05	186.9	7.944E+04
6513.51	21.15	187.3	7.944E+04
6812.64	21.28	187.7	7.944E+04
7121.07	21.41	188.1	7.944E+04
7428.60	21.54	188.5	7.943E+04
7738.67	21.63	188.8	7.944E+04
8035.10	21.72	189.1	7.943E+04
8346.39	21.78	189.4	7.944E+04
8658.07	21.85	189.7	7.945E+04
8970.17	21.91	190.0	7.946E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
9282.26	21.96	190.3	7.946E+04
9566.20	21.99	190.5	7.947E+04
9878.70	22.05	190.8	7.947E+04
10190.79	22.09	191.0	7.948E+04
10503.29	22.13	191.2	7.948E+04
10815.39	22.16	191.4	7.948E+04
11439.98	22.22	191.8	7.950E+04
12039.95	22.29	192.1	7.950E+04
12664.95	22.36	192.5	7.952E+04
13289.95	22.41	192.8	7.952E+04
13914.54	22.45	193.1	7.952E+04
14539.54	22.51	193.3	7.953E+04
15125.17	22.55	193.6	7.954E+04
15750.17	22.58	193.8	7.954E+04
16375.17	22.61	194.0	7.954E+04
17000.17	22.64	194.1	7.953E+04
17625.17	22.67	194.3	7.953E+04
18250.17	22.68	194.4	7.953E+04
18837.64	22.70	194.5	7.953E+04
19438.67	22.71	194.5	7.952E+04
20063.67	22.71	194.6	7.952E+04
20688.67	22.71	194.6	7.951E+04
21313.67	22.70	194.7	7.950E+04
21938.67	22.70	194.7	7.949E+04
22563.67	22.69	194.7	7.949E+04
23159.17	22.70	194.7	7.948E+04
23735.20	22.74	194.7	7.947E+04
24093.80		194.7	
24359.79	22.74	194.7	7.947E+04
24982.76	22.74	194.7	7.945E+04
25604.11	22.78	194.6	7.944E+04
26223.82	22.85	194.6	7.942E+04
26841.92	22.90	194.6	7.940E+04
27465.70	22.90	194.5	7.939E+04
28090.70	22.83	194.5	7.938E+04
28715.70	22.76	194.5	7.937E+04
29319.45	22.73	194.5	7.936E+04
29855.07	22.71	194.4	7.936E+04
30390.76	22.69	194.4	7.935E+04
30926.64	22.68	194.3	7.934E+04
31462.26	22.67	194.3	7.934E+04

Peak Suppression Pool Temperature

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
31997.89	22.65	194.2	7.934E+04
32533.76	22.63	194.1	7.933E+04
33069.45	22.61	194.0	7.931E+04
33605.07	22.60	193.9	7.931E+04
34140.95	22.58	193.8	7.930E+04
34676.57	22.56	193.7	7.929E+04
35212.20	22.54	193.6	7.928E+04
35748.07	22.51	193.5	7.928E+04
36283.76	22.49	193.4	7.927E+04
36819.39	22.47	193.3	7.926E+04
37355.07	22.45	193.2	7.925E+04
37890.95	22.42	193.0	7.924E+04
38426.57	22.40	192.9	7.923E+04
38961.14	22.38	192.8	7.922E+04
39497.01	22.34	192.6	7.921E+04
40032.64	22.31	192.5	7.920E+04
40568.26	22.28	192.3	7.919E+04
41104.14	22.25	192.2	7.918E+04
41639.82	22.22	192.0	7.917E+04
42175.45	22.19	191.8	7.916E+04
42711.14	22.16	191.7	7.915E+04
43247.01	22.13	191.5	7.915E+04
43782.64	22.09	191.4	7.914E+04
44318.26	22.06	191.2	7.913E+04
44854.14	22.03	191.0	7.912E+04
45400.32	22.01	190.9	7.911E+04
46025.32	21.96	190.7	7.909E+04
46650.32	21.93	190.6	7.908E+04
47275.32	21.90	190.4	7.907E+04
47900.32	21.87	190.3	7.907E+04
48525.32	21.84	190.1	7.905E+04
49150.32	21.81	189.9	7.904E+04
49775.32	21.77	189.8	7.903E+04
51025.32	21.71	189.5	7.901E+04
52275.32	21.65	189.1	7.900E+04
53525.32	21.59	188.8	7.898E+04
54772.64	21.52	188.5	7.896E+04
56022.64	21.46	188.1	7.895E+04
57272.64	21.41	187.8	7.893E+04
58522.64	21.35	187.4	7.891E+04
59772.64	21.28	187.1	7.890E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
61022.64	21.22	186.7	7.888E+04
62272.64	21.16	186.4	7.886E+04
63390.07	21.10	186.0	7.885E+04
64461.64	21.04	185.7	7.883E+04
65699.13	20.98	185.3	7.881E+04
66949.13	20.92	185.0	7.880E+04
68199.13	20.86	184.6	7.879E+04
69449.13	20.80	184.3	7.878E+04
70699.13	20.74	183.9	7.877E+04
71949.13	20.68	183.6	7.876E+04
73199.13	20.63	183.3	7.875E+04
74449.13	20.57	182.9	7.875E+04
75699.13	20.52	182.6	7.874E+04
76949.13	20.46	182.3	7.874E+04
78199.13	20.41	182.0	7.873E+04
79449.13	20.36	181.6	7.873E+04
80699.13	20.30	181.3	7.872E+04
81949.13	20.25	181.0	7.872E+04
83197.01	20.21	180.7	7.871E+04
84447.01	20.16	180.4	7.870E+04
85697.01	20.12	180.1	7.870E+04
86947.01	20.07	179.7	7.870E+04
88197.01	20.02	179.4	7.869E+04
89447.01	19.97	179.1	7.870E+04
90697.01	19.93	178.8	7.870E+04
91947.01	19.88	178.5	7.870E+04
93197.01	19.83	178.2	7.869E+04
94447.01	19.79	177.9	7.869E+04
95697.01	19.74	177.6	7.870E+04
96947.01	19.70	177.3	7.870E+04
98197.01	19.65	177.0	7.870E+04
99447.01	19.61	176.7	7.870E+04
100697.01	19.56	176.4	7.870E+04
101947.01	19.52	176.1	7.871E+04
103197.01	19.48	175.8	7.872E+04
104447.01	19.44	175.6	7.872E+04
105697.01	19.40	175.3	7.872E+04
106947.01	19.36	175.0	7.872E+04
108197.01	19.33	174.8	7.871E+04
109447.01	19.29	174.6	7.870E+04
110697.01	19.26	174.4	7.868E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
111947.01	19.23	174.2	7.866E+04
113197.01	19.19	174.0	7.864E+04
114447.01	19.16	173.8	7.860E+04
115697.01	19.12	173.6	7.858E+04
116947.01	19.09	173.4	7.855E+04
118197.01	19.06	173.2	7.852E+04
119447.01	19.03	173.1	7.849E+04
120697.01	19.00	172.9	7.846E+04
121942.70	18.97	172.7	7.843E+04
123192.70	18.94	172.5	7.841E+04
124442.70	18.91	172.4	7.838E+04
125692.70	18.88	172.2	7.835E+04
126934.82	18.85	172.0	7.832E+04
128184.82	18.83	171.9	7.829E+04
129434.82	18.80	171.7	7.827E+04
130684.82	18.77	171.5	7.824E+04
131934.83	18.74	171.4	7.822E+04
133184.83	18.72	171.2	7.819E+04
134434.83	18.70	171.0	7.816E+04
135684.83	18.67	170.9	7.814E+04
136934.83	18.64	170.7	7.811E+04
138184.83	18.62	170.6	7.809E+04
139434.83	18.59	170.4	7.806E+04
140684.83	18.56	170.2	7.803E+04
141934.83	18.54	170.1	7.801E+04
143184.83	18.51	169.9	7.799E+04
144434.83	18.49	169.8	7.797E+04
145684.83	18.46	169.6	7.793E+04
146934.83	18.44	169.4	7.791E+04
148184.83	18.41	169.3	7.789E+04
149434.83	18.39	169.1	7.787E+04
150684.83	18.36	169.0	7.785E+04
151934.83	18.34	168.8	7.782E+04
153184.83	18.31	168.7	7.780E+04
154434.83	18.29	168.5	7.777E+04
155684.83	18.26	168.3	7.775E+04
156934.83	18.24	168.2	7.773E+04
158184.83	18.22	168.0	7.771E+04
159434.83	18.19	167.9	7.769E+04
160684.83	18.17	167.7	7.767E+04
161934.83	18.14	167.6	7.765E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
163184.83	18.12	167.4	7.762E+04
164434.83	18.10	167.3	7.760E+04
165684.83	18.08	167.1	7.758E+04
166934.83	18.07	167.0	7.756E+04
168184.83	18.05	166.8	7.754E+04
169434.83	18.03	166.7	7.752E+04
170684.83	18.01	166.5	7.750E+04
171934.83	17.98	166.4	7.748E+04
173184.83	17.96	166.2	7.746E+04
174434.83	17.94	166.1	7.744E+04
175684.83	17.92	165.9	7.742E+04
176934.83	17.89	165.7	7.740E+04
178184.83	17.87	165.6	7.739E+04
179434.83	17.85	165.4	7.737E+04
180684.83	17.83	165.3	7.735E+04
181934.83	17.80	165.1	7.732E+04
183184.83	17.78	165.0	7.731E+04
184434.83	17.76	164.8	7.729E+04
185684.83	17.74	164.7	7.727E+04
186934.83	17.71	164.5	7.726E+04
188184.83	17.69	164.4	7.724E+04
189434.83	17.67	164.2	7.722E+04
190684.83	17.65	164.1	7.721E+04
191934.83	17.63	163.9	7.719E+04
193184.83	17.60	163.8	7.717E+04
194434.83	17.58	163.6	7.715E+04
195684.83	17.56	163.5	7.713E+04
196934.83	17.54	163.3	7.712E+04
198184.83	17.52	163.2	7.710E+04
199434.83	17.50	163.0	7.709E+04
200684.83	17.47	162.9	7.707E+04
201934.83	17.45	162.7	7.706E+04
203184.83	17.43	162.6	7.704E+04
204434.83	17.41	162.4	7.703E+04
205684.83	17.39	162.3	7.701E+04
206934.83	17.37	162.2	7.700E+04
208184.83	17.35	162.0	7.699E+04
209434.83	17.34	161.9	7.697E+04
210684.83	17.32	161.8	7.696E+04
211934.83	17.30	161.6	7.695E+04
213184.83	17.28	161.5	7.694E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
214434.83	17.26	161.4	7.691E+04
215684.83	17.25	161.3	7.690E+04
216934.83	17.23	161.1	7.689E+04
218184.83	17.21	161.0	7.688E+04
219434.83	17.20	160.9	7.686E+04
220684.83	17.18	160.8	7.685E+04
221934.83	17.17	160.7	7.684E+04
223184.83	17.15	160.6	7.683E+04
224434.83	17.13	160.5	7.682E+04
225684.83	17.12	160.4	7.681E+04
226934.83	17.10	160.3	7.680E+04
228184.83	17.09	160.2	7.679E+04
229434.83	17.07	160.1	7.678E+04
230684.83	17.06	160.0	7.676E+04
231934.83	17.05	159.9	7.675E+04
233184.83	17.03	159.8	7.674E+04
234434.83	17.02	159.7	7.673E+04
235684.83	17.00	159.6	7.671E+04
236934.83	16.99	159.5	7.670E+04
238184.83	16.98	159.4	7.669E+04
239434.83	16.96	159.3	7.668E+04
240684.83	16.95	159.2	7.667E+04
241934.83	16.94	159.1	7.666E+04
243184.83	16.92	159.0	7.666E+04
244434.83	16.91	158.9	7.665E+04
245684.83	16.90	158.8	7.664E+04
246934.83	16.88	158.8	7.663E+04
248184.83	16.87	158.7	7.662E+04
249434.83	16.86	158.6	7.661E+04
250684.83	16.85	158.5	7.660E+04
251934.83	16.83	158.4	7.659E+04
253183.83	16.82	158.3	7.659E+04
254433.83	16.81	158.3	7.658E+04
255683.83	16.80	158.2	7.657E+04
256933.83	16.79	158.1	7.656E+04
258183.83	16.77	158.0	7.655E+04
259433.83	16.76	157.9	7.655E+04
260683.83	16.75	157.8	7.654E+04
261933.83	16.74	157.8	7.653E+04
263183.81	16.73	157.7	7.652E+04
264433.81	16.72	157.6	7.651E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
265683.81	16.70	157.5	7.651E+04
266933.81	16.69	157.4	7.650E+04
268183.81	16.68	157.4	7.649E+04
269433.81	16.67	157.3	7.648E+04
270683.81	16.66	157.2	7.648E+04
271933.81	16.65	157.1	7.647E+04
273183.81	16.64	157.1	7.646E+04
274433.81	16.62	157.0	7.645E+04
275683.81	16.61	156.9	7.645E+04
276933.81	16.60	156.8	7.644E+04
278183.81	16.59	156.7	7.643E+04
279433.81	16.58	156.7	7.643E+04
280683.81	16.57	156.6	7.642E+04
281933.81	16.56	156.5	7.641E+04
283183.81	16.55	156.5	7.641E+04
284433.81	16.54	156.4	7.640E+04
285683.81	16.52	156.3	7.640E+04
286933.81	16.50	156.1	7.638E+04
288183.81	16.49	155.9	7.638E+04
289433.81	16.47	155.8	7.637E+04
290683.81	16.45	155.7	7.636E+04
291933.81	16.44	155.5	7.635E+04
293183.81	16.42	155.4	7.635E+04
294433.81	16.40	155.3	7.634E+04
295683.81	16.39	155.2	7.633E+04
296933.81	16.37	155.0	7.633E+04
298183.81	16.36	154.9	7.632E+04
299433.81	16.34	154.8	7.631E+04
300683.81	16.33	154.7	7.631E+04
301933.81	16.32	154.6	7.630E+04
303183.81	16.30	154.4	7.630E+04
304433.81	16.29	154.3	7.629E+04
305683.81	16.28	154.2	7.630E+04
306933.81	16.26	154.1	7.629E+04
308183.81	16.25	154.0	7.629E+04
309433.81	16.24	153.9	7.628E+04
310683.81	16.22	153.8	7.628E+04
311933.81	16.21	153.7	7.627E+04
313183.81	16.20	153.6	7.627E+04
314433.81	16.19	153.5	7.626E+04
315683.81	16.17	153.4	7.626E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
316933.81	16.16	153.3	7.625E+04
318183.81	16.15	153.2	7.625E+04
319433.81	16.14	153.1	7.625E+04
320683.81	16.13	153.0	7.624E+04
321933.81	16.11	152.9	7.624E+04
323183.81	16.10	152.8	7.623E+04
324433.81	16.09	152.7	7.623E+04
325683.81	16.08	152.7	7.623E+04
326933.81	16.07	152.6	7.622E+04
328183.81	16.06	152.5	7.622E+04
329433.81	16.05	152.4	7.622E+04
330683.81	16.04	152.3	7.621E+04
331933.81	16.03	152.2	7.621E+04
333183.81	16.02	152.1	7.621E+04
334433.81	16.01	152.0	7.621E+04
335683.81	16.00	152.0	7.620E+04
336933.81	15.98	151.9	7.620E+04
338183.81	15.97	151.8	7.620E+04
339433.81	15.96	151.7	7.619E+04
340683.81	15.95	151.6	7.619E+04
341933.81	15.94	151.5	7.619E+04
343183.81	15.93	151.5	7.619E+04
344433.81	15.92	151.4	7.620E+04
345683.81	15.91	151.3	7.619E+04
346933.81	15.90	151.2	7.619E+04
348183.81	15.89	151.1	7.619E+04
349433.81	15.88	151.1	7.619E+04
350683.81	15.87	151.0	7.619E+04
351933.81	15.86	150.9	7.618E+04
353183.81	15.85	150.8	7.618E+04
354433.81	15.84	150.7	7.618E+04
355683.81	15.83	150.7	7.618E+04
356933.81	15.82	150.6	7.618E+04
358183.81	15.82	150.5	7.618E+04
359433.81	15.81	150.4	7.618E+04
360683.81	15.80	150.3	7.618E+04
361933.81	15.79	150.3	7.617E+04
363183.81	15.78	150.2	7.618E+04
364433.81	15.77	150.1	7.618E+04
365683.81	15.76	150.0	7.618E+04
366933.81	15.75	150.0	7.618E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
368183.81	15.74	149.9	7.618E+04
369433.81	15.73	149.8	7.618E+04
370683.81	15.72	149.7	7.618E+04
371933.81	15.71	149.7	7.618E+04
373183.81	15.70	149.6	7.618E+04
374433.81	15.70	149.5	7.618E+04
375683.81	15.69	149.4	7.618E+04
376933.81	15.68	149.4	7.618E+04
378183.81	15.67	149.3	7.618E+04
379433.81	15.66	149.2	7.619E+04
380683.81	15.65	149.1	7.619E+04
381933.81	15.64	149.1	7.619E+04
383183.81	15.63	149.0	7.619E+04
384433.81	15.63	148.9	7.619E+04
385683.81	15.62	148.8	7.619E+04
386933.81	15.61	148.8	7.619E+04
388183.81	15.60	148.7	7.619E+04
389433.81	15.59	148.6	7.619E+04
390683.81	15.57	148.6	7.619E+04
391933.81	15.57	148.5	7.619E+04
393183.81	15.56	148.4	7.619E+04
394433.81	15.55	148.3	7.619E+04
395683.81	15.53	148.3	7.619E+04
396933.81	15.53	148.2	7.620E+04
398183.81	15.53	148.1	7.620E+04
399433.81	15.52	148.0	7.620E+04
400683.81	15.50	148.0	7.620E+04
401933.81	15.50	147.9	7.620E+04
403183.81	15.49	147.8	7.620E+04
404433.81	15.48	147.8	7.620E+04
405683.81	15.47	147.7	7.622E+04
406933.81	15.47	147.6	7.621E+04
408183.81	15.46	147.6	7.621E+04
409433.81	15.45	147.5	7.621E+04
410683.81	15.43	147.5	7.622E+04
411933.81	15.43	147.4	7.622E+04
413183.81	15.43	147.4	7.622E+04
414433.81	15.42	147.3	7.622E+04
415683.81	15.40	147.2	7.624E+04
416933.81	15.40	147.2	7.623E+04
418183.81	15.40	147.1	7.623E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
419433.81	15.39	147.1	7.624E+04
420683.81	15.38	147.0	7.625E+04
421933.81	15.38	147.0	7.624E+04
423183.81	15.37	146.9	7.624E+04
424433.81	15.37	146.9	7.625E+04
425683.81	15.35	146.9	7.626E+04
426933.81	15.35	146.8	7.626E+04
428183.81	15.35	146.8	7.626E+04
429433.81	15.32	146.7	7.626E+04
430683.81	15.33	146.7	7.627E+04
431933.81	15.33	146.6	7.627E+04
433183.81	15.32	146.6	7.627E+04
434433.81	15.30	146.6	7.628E+04
435683.81	15.30	146.5	7.628E+04
436933.81	15.30	146.5	7.628E+04
438183.81	15.30	146.4	7.629E+04
439434.06	15.28	146.4	7.630E+04
440684.06	15.28	146.4	7.630E+04
441934.06	15.28	146.3	7.631E+04
443181.56	15.27	146.3	7.630E+04
444432.31	15.25	146.3	7.630E+04
445682.31	15.26	146.2	7.631E+04
446932.31	15.26	146.2	7.632E+04
448182.31	15.25	146.1	7.633E+04
449432.31	15.23	146.1	7.633E+04
450682.31	15.24	146.1	7.633E+04
451932.31	15.24	146.0	7.633E+04
453182.31	15.23	146.0	7.633E+04
454432.44	15.22	146.0	7.634E+04
455682.44	15.22	145.9	7.634E+04
456932.44	15.22	145.9	7.636E+04
458182.81	15.20	145.9	7.636E+04
459432.81	15.20	145.9	7.636E+04
460682.81	15.20	145.8	7.637E+04
461932.81	15.19	145.8	7.637E+04
463182.94	15.18	145.8	7.638E+04
464432.94	15.18	145.7	7.638E+04
465683.06	15.17	145.7	7.639E+04
466933.06	15.17	145.7	7.639E+04
468183.06	15.17	145.6	7.640E+04
469433.06	15.17	145.6	7.640E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
470683.06	15.16	145.6	7.640E+04
471933.31	15.15	145.6	7.640E+04
473183.31	15.15	145.6	7.640E+04
474433.31	15.15	145.6	7.640E+04
475683.31	15.15	145.6	7.641E+04
476933.81	15.14	145.5	7.641E+04
478183.81	15.13	145.5	7.641E+04
479433.81	15.13	145.5	7.642E+04
480683.81	15.13	145.5	7.642E+04
481934.06	15.11	145.5	7.642E+04
483184.06	15.12	145.4	7.642E+04
484434.06	15.11	145.4	7.643E+04
485684.69	15.10	145.4	7.643E+04
486934.69	15.10	145.3	7.644E+04
488184.69	15.10	145.3	7.645E+04
489434.69	15.10	145.3	7.645E+04
490684.69	15.09	145.3	7.645E+04
491935.44	15.08	145.2	7.646E+04
493185.44	15.08	145.2	7.646E+04
494435.44	15.08	145.2	7.646E+04
495685.44	15.07	145.2	7.646E+04
496935.69	15.06	145.2	7.647E+04
498185.69	15.06	145.1	7.648E+04
499435.69	15.05	145.1	7.648E+04
500685.69	15.05	145.1	7.648E+04
501935.69	15.05	145.1	7.648E+04
503185.69	15.05	145.0	7.648E+04
504435.81	15.03	145.0	7.648E+04
505689.50	15.03	145.0	7.649E+04
506939.50	15.03	145.0	7.649E+04
508189.50	15.03	144.9	7.650E+04
509439.50	15.02	144.9	7.650E+04
510689.50	15.01	144.9	7.650E+04
511939.50	15.01	144.9	7.651E+04
513189.50	15.00	144.9	7.651E+04
514439.50	15.00	144.8	7.651E+04
515689.50	15.00	144.8	7.651E+04
516939.50	14.99	144.8	7.653E+04
518189.50	14.98	144.7	7.653E+04
519439.50	14.98	144.7	7.653E+04
520689.50	14.98	144.7	7.653E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
521939.50	14.98	144.7	7.653E+04
523189.63	14.96	144.6	7.654E+04
524439.63	14.96	144.6	7.654E+04
525689.63	14.96	144.6	7.654E+04
526939.63	14.95	144.6	7.655E+04
528189.63	14.95	144.6	7.655E+04
529439.63	14.95	144.5	7.655E+04
530689.63	14.93	144.5	7.656E+04
531939.63	14.93	144.5	7.656E+04
533189.63	14.93	144.5	7.657E+04
534439.63	14.93	144.5	7.657E+04
535689.63	14.92	144.4	7.657E+04
536939.63	14.92	144.4	7.657E+04
538189.63	14.91	144.4	7.658E+04
539439.63	14.91	144.3	7.658E+04
540689.63	14.90	144.3	7.659E+04
541939.63	14.90	144.3	7.659E+04
543189.63	14.89	144.2	7.660E+04
544439.63	14.88	144.2	7.660E+04
545689.63	14.88	144.2	7.661E+04
546939.63	14.88	144.1	7.661E+04
548189.63	14.87	144.1	7.662E+04
549439.63	14.86	144.1	7.662E+04
550689.63	14.86	144.0	7.663E+04
551939.63	14.86	144.0	7.663E+04
553189.75	14.84	144.0	7.664E+04
554439.75	14.84	144.0	7.664E+04
555689.75	14.84	143.9	7.665E+04
556939.75	14.84	143.9	7.665E+04
558190.00	14.82	143.9	7.666E+04
559440.00	14.82	143.8	7.666E+04
560690.00	14.82	143.8	7.666E+04
561940.00	14.82	143.8	7.667E+04
563190.13	14.80	143.7	7.668E+04
564440.13	14.80	143.7	7.668E+04
565690.13	14.80	143.7	7.668E+04
566940.13	14.80	143.6	7.669E+04
568190.38	14.78	143.6	7.670E+04
569440.38	14.78	143.6	7.670E+04
570690.38	14.78	143.6	7.671E+04
571940.38	14.78	143.5	7.671E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
573190.38	14.77	143.5	7.672E+04
574440.50	14.76	143.5	7.673E+04
575690.50	14.76	143.5	7.673E+04
576940.50	14.76	143.5	7.674E+04
578190.50	14.75	143.5	7.675E+04
579440.50	14.75	143.5	7.675E+04
580690.50	14.75	143.5	7.676E+04
581940.50	14.75	143.5	7.677E+04
583190.50	14.75	143.5	7.677E+04
584440.50	14.74	143.5	7.679E+04
585690.50	14.74	143.5	7.679E+04
586940.50	14.74	143.5	7.680E+04
588190.50	14.74	143.5	7.680E+04
589440.50	14.74	143.5	7.680E+04
590690.50	14.73	143.5	7.681E+04
591940.50	14.73	143.5	7.682E+04
593190.50	14.72	143.4	7.682E+04
594440.50	14.71	143.4	7.683E+04
595690.50	14.71	143.3	7.683E+04
596940.50	14.70	143.3	7.684E+04
598190.50	14.70	143.3	7.684E+04
599440.50	14.70	143.2	7.684E+04
600690.50	14.70	143.2	7.686E+04
601940.50	14.70	143.1	7.686E+04
603190.50	14.70	143.1	7.686E+04
604440.50	14.70	143.1	7.687E+04
605690.50	14.70	143.0	7.688E+04
606940.50	14.70	143.0	7.688E+04
608190.50	14.70	143.0	7.688E+04
609440.50	14.70	143.0	7.689E+04
610690.50	14.70	143.0	7.689E+04
611940.50	14.70	142.9	7.689E+04
613190.50	14.70	142.9	7.689E+04
614440.50	14.70	142.9	7.690E+04
615690.50	14.70	142.8	7.691E+04
616940.50	14.70	142.8	7.691E+04
618190.50	14.70	142.8	7.691E+04
619440.50	14.70	142.8	7.693E+04
620690.50	14.70	142.7	7.693E+04
621940.50	14.70	142.7	7.693E+04
623190.50	14.70	142.6	7.694E+04

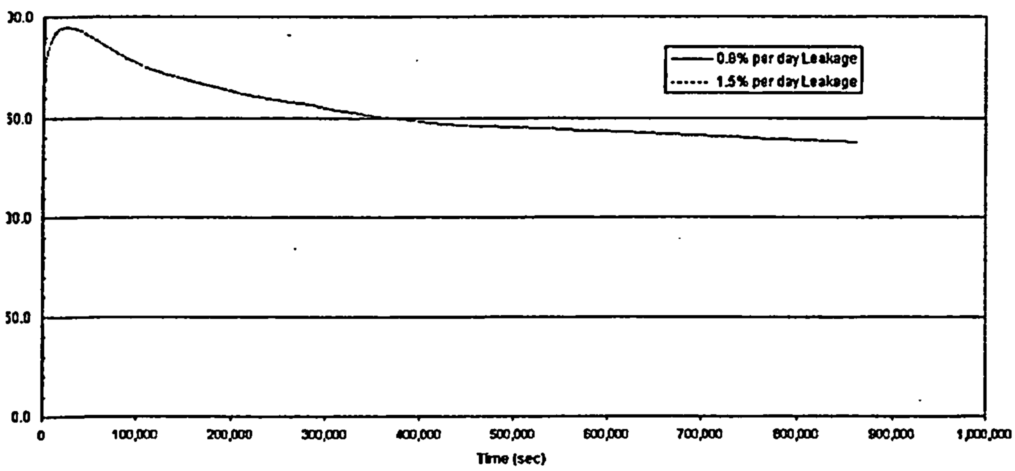
Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
624440.50	14.70	142.6	7.695E+04
625690.50	14.70	142.6	7.695E+04
626940.50	14.70	142.6	7.695E+04
628190.50	14.70	142.5	7.696E+04
629440.50	14.70	142.5	7.697E+04
630690.50	14.70	142.5	7.697E+04
631940.50	14.70	142.5	7.697E+04
633190.50	14.70	142.4	7.697E+04
634440.50	14.70	142.4	7.697E+04
635690.50	14.70	142.4	7.698E+04
636940.50	14.70	142.4	7.699E+04
638190.50	14.70	142.3	7.699E+04
639440.50	14.70	142.3	7.701E+04
640690.50	14.70	142.3	7.700E+04
641940.50	14.70	142.3	7.701E+04
643190.50	14.70	142.2	7.701E+04
644440.50	14.70	142.2	7.702E+04
645690.50	14.70	142.2	7.703E+04
646940.50	14.70	142.1	7.703E+04
648190.50	14.70	142.1	7.703E+04
649440.50	14.70	142.1	7.705E+04
650690.50	14.70	142.0	7.705E+04
651940.50	14.70	142.0	7.706E+04
653190.50	14.70	142.0	7.707E+04
654440.50	14.70	141.9	7.707E+04
655690.50	14.70	141.9	7.708E+04
656940.50	14.70	141.9	7.708E+04
658190.50	14.70	141.9	7.709E+04
659440.50	14.70	141.8	7.710E+04
660690.50	14.70	141.8	7.710E+04
661940.50	14.70	141.8	7.711E+04
663190.50	14.70	141.7	7.712E+04
664440.50	14.70	141.7	7.712E+04
665690.50	14.70	141.7	7.712E+04
666940.50	14.70	141.7	7.713E+04
668190.50	14.70	141.6	7.714E+04
669440.50	14.70	141.6	7.714E+04
670690.50	14.70	141.6	7.715E+04
671940.50	14.70	141.5	7.716E+04
673190.50	14.70	141.5	7.716E+04
674440.50	14.70	141.5	7.716E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
675690.50	14.70	141.5	7.717E+04
676940.50	14.70	141.5	7.718E+04
678190.50	14.70	141.4	7.718E+04
679440.50	14.70	141.4	7.718E+04
680690.50	14.70	141.4	7.719E+04
681940.50	14.70	141.4	7.719E+04
683190.50	14.70	141.4	7.720E+04
684440.50	14.70	141.3	7.721E+04
685690.50	14.70	141.3	7.721E+04
686940.50	14.70	141.3	7.722E+04
688190.50	14.70	141.2	7.723E+04
689440.50	14.70	141.2	7.723E+04
690690.50	14.70	141.2	7.724E+04
691940.50	14.70	141.1	7.725E+04
693190.50	14.70	141.1	7.725E+04
694440.50	14.70	141.1	7.725E+04
695690.50	14.70	141.1	7.725E+04
696940.50	14.70	141.1	7.726E+04
698190.50	14.70	141.1	7.727E+04
699440.50	14.70	141.0	7.727E+04
700690.50	14.70	141.0	7.728E+04
701940.50	14.70	141.0	7.729E+04
703190.50	14.70	140.9	7.729E+04
704440.50	14.70	140.9	7.730E+04
705690.50	14.70	140.9	7.731E+04
706940.50	14.70	140.8	7.731E+04
708190.50	14.70	140.8	7.732E+04
709440.50	14.70	140.8	7.732E+04
710690.50	14.70	140.8	7.733E+04
711940.50	14.70	140.8	7.733E+04
713190.50	14.70	140.8	7.733E+04
714440.50	14.70	140.8	7.734E+04
715690.50	14.70	140.7	7.734E+04
716940.50	14.70	140.7	7.734E+04
718190.50	14.70	140.7	7.736E+04
719440.50	14.70	140.6	7.736E+04
720690.50	14.70	140.6	7.737E+04
721940.50	14.70	140.6	7.738E+04
723190.50	14.70	140.6	7.738E+04
724440.50	14.70	140.6	7.738E+04
725690.50	14.70	140.5	7.738E+04

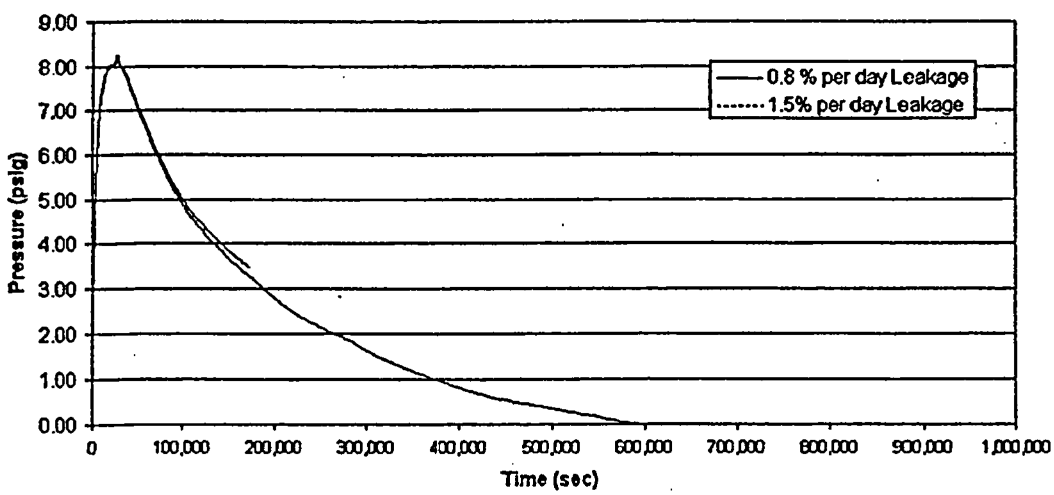
Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
726940.50	14.70	140.5	7.739E+04
728190.50	14.70	140.5	7.740E+04
729440.50	14.70	140.4	7.740E+04
730690.50	14.70	140.4	7.741E+04
731940.50	14.70	140.4	7.742E+04
733190.50	14.70	140.4	7.742E+04
734440.50	14.70	140.4	7.742E+04
735690.50	14.70	140.4	7.742E+04
736940.50	14.70	140.3	7.744E+04
738190.50	14.70	140.3	7.744E+04
739440.50	14.70	140.3	7.745E+04
740690.50	14.70	140.2	7.746E+04
741940.50	14.70	140.2	7.746E+04
743190.50	14.70	140.2	7.746E+04
744440.50	14.70	140.2	7.746E+04
745690.50	14.70	140.2	7.747E+04
746940.50	14.70	140.1	7.748E+04
748190.50	14.70	140.1	7.748E+04
749440.50	14.70	140.1	7.749E+04
750690.50	14.70	140.0	7.750E+04
751940.50	14.70	140.0	7.751E+04
753190.50	14.70	140.0	7.751E+04
754440.50	14.70	139.9	7.752E+04
755690.50	14.70	139.9	7.753E+04
756940.50	14.70	139.9	7.753E+04
758190.50	14.70	139.8	7.755E+04
759440.50	14.70	139.8	7.755E+04
760690.50	14.70	139.8	7.757E+04
761940.50	14.70	139.7	7.757E+04
763190.50	14.70	139.7	7.758E+04
764440.50	14.70	139.7	7.759E+04
765690.50	14.70	139.6	7.759E+04
766940.50	14.70	139.6	7.760E+04
768190.50	14.70	139.6	7.761E+04
769440.50	14.70	139.6	7.761E+04
770690.50	14.70	139.6	7.762E+04
771940.50	14.70	139.5	7.762E+04
773190.50	14.70	139.5	7.762E+04
774440.50	14.70	139.5	7.763E+04
775690.50	14.70	139.5	7.764E+04
776940.50	14.70	139.4	7.764E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
778190.50	14.70	139.4	7.766E+04
779440.50	14.70	139.4	7.766E+04
780690.50	14.70	139.4	7.766E+04
781940.50	14.70	139.3	7.768E+04
783190.50	14.70	139.3	7.768E+04
784440.50	14.70	139.3	7.768E+04
785690.50	14.70	139.3	7.768E+04
786940.50	14.70	139.3	7.769E+04
788190.50	14.70	139.3	7.770E+04
789440.50	14.70	139.2	7.770E+04
790690.50	14.70	139.2	7.770E+04
791940.50	14.70	139.2	7.772E+04
793190.50	14.70	139.2	7.772E+04
794440.50	14.70	139.1	7.772E+04
795690.50	14.70	139.1	7.773E+04
796940.50	14.70	139.1	7.774E+04
798190.50	14.70	139.0	7.775E+04
799440.50	14.70	139.0	7.775E+04
800690.50	14.70	139.0	7.777E+04
801940.50	14.70	138.9	7.777E+04
803190.50	14.70	138.9	7.777E+04
804440.50	14.70	138.9	7.777E+04
805690.50	14.70	138.9	7.779E+04
806940.50	14.70	138.9	7.779E+04
808190.50	14.70	138.8	7.780E+04
809440.50	14.70	138.8	7.781E+04
810690.50	14.70	138.8	7.781E+04
811940.50	14.70	138.8	7.781E+04
813190.50	14.70	138.8	7.782E+04
814440.50	14.70	138.7	7.783E+04
815690.50	14.70	138.7	7.783E+04
816940.50	14.70	138.7	7.784E+04
818190.50	14.70	138.7	7.784E+04
819440.50	14.70	138.6	7.785E+04
820690.50	14.70	138.6	7.786E+04
821940.50	14.70	138.6	7.786E+04
823190.50	14.70	138.5	7.788E+04
824440.50	14.70	138.5	7.788E+04
825690.50	14.70	138.5	7.789E+04
826940.50	14.70	138.5	7.790E+04
828190.50	14.70	138.4	7.790E+04

Time (seconds)	Wetwell Pressure (psia)	Suppression Pool Temperature (°F)	Suppression Pool Volume (ft^3)
829440.50	14.70	138.4	7.791E+04
830690.50	14.70	138.4	7.792E+04
831940.50	14.70	138.3	7.793E+04
833190.50	14.70	138.3	7.793E+04
834440.50	14.70	138.3	7.795E+04
835690.50	14.70	138.3	7.795E+04
836940.50	14.70	138.3	7.795E+04
838190.50	14.70	138.2	7.795E+04
839440.50	14.70	138.2	7.795E+04
840690.50	14.70	138.2	7.795E+04
841940.50	14.70	138.2	7.797E+04
843190.50	14.70	138.2	7.797E+04
844440.50	14.70	138.2	7.797E+04
845690.50	14.70	138.2	7.797E+04
846940.50	14.70	138.1	7.799E+04
848190.50	14.70	138.1	7.799E+04
849440.50	14.70	138.1	7.799E+04
850690.50	14.70	138.1	7.799E+04
851940.50	14.70	138.1	7.801E+04
853190.50	14.70	138.0	7.801E+04
854440.50	14.70	138.0	7.802E+04
855690.50	14.70	137.9	7.803E+04
856940.50	14.70	137.9	7.804E+04
858190.50	14.70	137.9	7.805E+04
859440.50	14.70	137.9	7.805E+04
860690.50	14.70	137.8	7.806E+04
861940.50	14.70	137.8	7.806E+04
863190.50	14.70	137.8	7.807E+04
864000.00	14.70	137.8	7.808E+04



Long-term DBA-LOCA NPSH Suppression Pool Temperature



Long-term DBA-LOCA NPSH Suppression Pool Pressure

minimum available NPSH requirements. The pump manufacturer's recommended minimum available NPSH is based on assuring acceptable pump performance and reliability. See the response to RAI SPSB-C-25 for additional information on pump performance and cavitation. The intent of Regulatory Position 2.1.1.3 is met by meeting the minimum NPSH requirement.

The treatment of decay and residual heat meets the intent of Regulatory Position 2.1.1.4. The calculations assume a 2% calorimetric uncertainty and the decay heat calculation also applies the ANSI/ANS 5.1-1979 2-sigma uncertainty. The decay heat assumptions are also discussed in the responses to RAIs SPSB-C-4 and SPSB-C-5.

Regulatory Position 2.1.1.5 states that the hot channel correlation factor specified in ANSI/HI 1.1-1.5-1994 should not be used. VYNPS has not applied this correlation in determining the margin between the available and required NPSH.

The initial suppression pool volume assumed in the suppression pool temperature and NPSH calculations is the Technical Specification minimum value. This assumption meets the intent of Regulatory Position 2.1.1.6.

Piping losses in the pump suction that affect the NPSH have been included in the calculation. This meets the intent of Regulatory Position 2.1.1.7. The response to RAI SPSB-C-26 includes the NPSH calculation.

The NPSH treatment of suction strainer screen debris loading meets the intent of Regulatory Position 2.1.1.8. The response to RAI SPSB-C-6 notes that there have been no changes adversely affecting debris loading since the completion of actions requested by NRC Bulletin 96-03.

Regulatory Position 2.1.1.9 states that NPSH calculations should be performed as a function of time until it is clear that the available NPSH will not decrease further. The NPSH calculation meets the intent of this regulatory position. Revised Figure 4-6, provided with the response to RAI SPSB-C-23, shows the time dependent NPSH requirements. The figure shows that NPSH will not decrease further as demonstrated by the decreasing amount of overpressure credit required after approximately eight hours.

RAI SPSB-C-22

Describe how the VYNPS emergency operating procedures will be revised to ensure that the containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH.

Response to RAI SPSB-C-22

The VYNPS emergency operating procedures (EOPs) do not require revision to ensure that the containment accident pressure will be prevented from falling below the pressure required for adequate available NPSH. Current EOPs incorporate guidance to ensure that containment

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accident pressure will be prevented from falling below the pressure required for adequate available NPSH.

Per VYNPS emergency operating procedure (EOP) EOP-1, "RPV Control," after an automatic action level has been reached, operators are directed to verify applicable automatic actions have occurred. Verifying automatic actions provides backup confirmation that all isolation valves have closed on a primary containment isolation signal.

VYNPS EOPs establish NPSH limits for residual heat removal (RHR) and core spray (CS) pumps. (Separate limits are provided for RHR and CS). The NPSH limit is a function of pump flow, torus water temperature, and suppression chamber pressure. It is used to preclude ECCS pump damage due to cavitation and to ensure adequate coolant flow. As overpressure increases, the static pressure and margin to saturation at the pump inlet also increase. The available NPSH therefore increases with overpressure.

In accordance with EOP-1, when using RHR for an injection system, operators are directed to inject through the heat exchanger as soon as possible and to control and maintain pump flow below the RHR NPSH Limit. For the core spray system, operators are directed to control and maintain pump flow below the CS NPSH Limit.

EOP-3, "Primary Containment Control," Note 5 states: "Reducing primary containment pressure will reduce the available NPSH for pumps taking suction from the torus." Per the EOP Study Guide, if there is no future need for sprays and containment overpressure is desired to provide adequate NPSH for pumps drawing suction from the suppression pool, sprays may be terminated at a higher pressure.

In accordance with EOP-3, drywell sprays are initiated before containment temperature reaches 280 °F or when torus pressure exceeds 10 psi. Containment sprays should isolate automatically when drywell pressure decreases to 2.5 psig. Both of these steps in EOP-3 provide reference to Caution #5 emphasizing the relationship between primary containment pressure and available NPSH.

Also, per EOP-3, once the high drywell pressure isolation occurs, containment venting is directed only after a reactor pressure vessel emergency depressurization (RPV-ED) is required and prior to exceeding the primary containment pressure limit (PCPL-A curve in EOP-3). In the event that containment venting is required, operators will vent the containment to control pressure below the PCPL-A curve. The pressure at which containment is maintained during venting is based on considerations of NPSH for the RHR and core spray pumps, expected release rates, and total releases. Therefore, sufficient containment overpressure is preserved.

RAI SPSB-C-23

The application dated September 10, 2003 (Reference 1), Attachment 6, Table 4-2 and Figure 4-6, show that the containment accident pressure requested for adequate available NPSH is 1.3 psig at 50 hours. When is containment accident pressure no longer required?