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Date: 9/17/04 10:34AM
Subject: Request to NRC ACRS

Yesterday, the Department requested that the NRC Advisory Committee on Reactor Safeguards (ACRS) review Vermont Yankee's request to change its design basis to take credit for containment overpressure to demonstrate the adequacy of emergency core cooling pumps. The ACRS is will already review Vermont Yankee's request to increase its power level by 20%. We asked ACRS to specifically focus on the overpressure credit issue.

Attached for your information is the Department's September 17, 2004 letter.

<<ACRSLtr9-17-04FINAL.pdf>> <<ACRS-Attachment A.pdf>> <<ACRS-Attachment B.pdf>> <<ACRS-Attachment C.pdf>> <<ACRS-Attachment D.pdf>>

C-18

September 17, 2004

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

RE: State of Vermont Request to Consider the Containment Overpressure Credit Policy

Dear Chairman Bonaca:

On behalf of the State of Vermont, by this letter I am requesting that the ACRS ("the Committee") specifically review, as part of Entergy's request for extended power uprate for the Vermont Yankee nuclear plant, Entergy's request to change Vermont Yankee's design basis to take credit for containment overpressure to demonstrate the adequacy of its emergency core cooling and containment spray pumps, and the NRC staff's ("the staff") policy of granting such requests. The reasons for the State of Vermont request are presented below.

We have reviewed Entergy's request for extended power uprate for Vermont Yankee. We question the prudence of removing the safety margin for the adequate functioning of emergency core cooling and containment cooling pumps provided by containment overpressure. That safety margin was established with good basis in the first Regulatory Guide, then Safety Guide 1.

From December 2003, we have conducted correspondence with the staff regarding the proposed removal of this important safety margin, but we have not yet received satisfactory answers to our questions, nor have we seen a convincing reason why such overpressure credit should be granted. Therefore, on August 30, 2004, we petitioned NRC for a hearing on the issue as it relates to Vermont Yankee. The technical contentions from our petition are included as Attachment A. Our letter to the staff of December 8, 2003, and the staff's answer on June 29, 2004 are provided as Attachment B and C, respectively.

At an early Vermont Yankee power uprate meeting at NRC headquarters, NRC staff asserted that the basis for considering containment overpressure credit was Regulatory Guide 1.82, Rev.

3 (then Draft Regulatory Guide 1107), which, in a very small portion, addresses the containment overpressure credit. I will now recount the history leading up to the overpressure credit statements in Regulatory Guide 1.82, Rev. 3. For most of this history, the NRC, industry and the Committee were adamant about retaining the defense-in-depth safety margin provided by not linking emergency core and containment cooling functions with containment performance. In the late-1990's, overpressure credit began to be granted selectively for a few cases based on need related to strainer debris loading. And then, with the advent of extended power uprates, it appears the NRC began granting overpressure credit whenever an Applicant asked for such credit.

- From the early 1970's, and before, until the mid-1990's, the principle expressed in Regulatory Guide 1.1 of not allowing containment overpressure credit for net positive suction head (NPSH) calculations seems to have been upheld.

- Vermont Yankee's design basis does not allow containment overpressure credit for demonstration of adequate NPSH.

- In the mid-1990's, as a result of ECCS suction strainer fouling at a number of boiling water reactors (BWRs), regulatory actions were taken to cause BWRs to modify suction strainers and recalculate the adequacy of NPSH for emergency core cooling and containment spray pumps. The Committee was highly involved in this review and Regulatory Guide 1.82, Rev. 2 (May 1996), was one of the regulatory products of this review[1].

- In 1996, all parties (industry, NRC staff, ACRS) agreed that overpressure credit should not be granted for NPSH calculations. Regulatory Guide 1.82, Rev. 2, is clear regarding no containment overpressure credit. Section 2.3.3.4 states:

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[1] In its recommendation of Regulatory Guide 1.82, Rev. 2, the Committee faulted the industry and the staff: "Finally, we continue to believe that the response of the staff and the BWR licensees to this important nuclear safety issue has been unacceptably slow." ACRS Letter of February 26, 1996, *Proposed Final NRC Bulletin 96-XX, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors" and an associated Draft Revision 2 of Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling following a Loss-of-Coolant Accident."*

> The NPSH available to the ECC pumps should be determined using conditions specified in the plant's licensing basis (e.g., Regulatory Guide 1.1).

The BWR strainer regulatory actions in the mid- and late-1990's are summarized by Los Alamos National Laboratory (LANL) in the NRC sponsored research paper, "*BWR ECCS Strainer Blockage Issue: Summary of Research and Resolution Actions,*" LA-UR-01-1595, March 21, 2001. The following are quotations from the LANL report regarding containment overpressure credit and uncertainty:

> Note that the BWROG does not recommend crediting containment overpressure in calculating NPSH margins. (page 4-2).
>
> The staff concurs that additional containment overpressure (other than an amount already approved by the staff for the existing licensing basis) should not be used as part of the resolution of this issue. (page 4-3).
>
> Some licensees discovered that they must take new credit for containment overpressure to meet the NPSH requirements of the ECCS and containment heat removal pumps and the overpressure being credited by licensees may be inconsistent with the plant's respective licensing basis. The staff further evaluated its position on use of containment overpressure in calculating NPSH margin and recommended that licensing basis changes not be used as a resolution option due to the substantial uncertainty associated with determining NPSH margin. (page x).
>
> The staff also noted that a good practice would be to maintain defense-in-depth because of the uncertainties associated with any resolution of this issue. (page 4-4).
>
> The Advisory Committee on Reactor Safeguards (ACRS) agreed with the BWROG. In a letter from the ACRS to the NRC Executive Director for Operations (EDO) explicitly stated, "We believe that allowing some level of containment

> overpressure is not an acceptable corrective action because adequate overpressure may not be present when needed." (page 1-14).
>
> However, due to incomplete guidance and inadequate supporting documentation or analysis in several areas, the staff was unable to determine if all of the methodologies, or combination of methodologies, were conservative. Similarly, much of the general guidance on "resolution options" also lacked sufficient detail for the staff to review. Since the staff lacked sufficient detail and supporting justification on many of the "resolution options," these were generally considered unacceptable without further supporting justification from a licensee or the BWROG. (page 4-2).

- Despite the 1996 determinations of industry, staff and the Committee, numbers of BWRs (but not Vermont Yankee) needed to rely on containment overpressure credit to demonstrate the adequacy of existing designs with revised ECCS strainer loadings. Accordingly, the NRC staff began granting this credit. The Committee concluded in its December 12, 1997 letter, *Credit for Containment Overpressure to Provide Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps*:

  > As a result of further review of this issue, we now concur with the NRC staff position that *selectively* granting credit for small amounts of overpressure for *a few cases* may be justified. (Emphasis added).

  This letter went on to express concern over the completeness of the staff's consideration for granting this credit and its assessment of risk probabilities.

- During the intervening eight years (between Rev. 2 and Rev. 3 of Regulatory Guide 1.82), the staff has granted overpressure credit to a number of plants and the expectation exists that PWRs will require containment overpressure credit as part of resolution of the pending PWR sump/strainer issues. Regulatory Guide 1.82, Rev. 3, as approved by the Committee, contains the following:

> **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. [Emphasis added.]

In addition, the introductory portion of Regulatory Guide 1.82, Rev. 3, contains the following statement, at 8:

> 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.]

- Also during the intervening eight years, the staff has considered and granted BWR extended power uprate amendments which also included grants of containment overpressure credit[2]. For NPSH adequacy, it appears the staff's willingness to grant selective overpressure credit for sump/strainer loading resolution, has been changed into a general grant of overpressure credit for power uprate. It appears it has now become standard policy to grant containment overpressure credit to extended power uprate applicants whenever asked.

[2] Extended power uprate creates more energy transfer during a postulated LOCA that results in higher sump (or torus) temperatures, in turn resulting in reduced available NPSH for ECCS and containment heat removal pumps.

Considering the history stated above, the State of Vermont believes the following:

1. We believe the staff has made a major policy change in reducing defense-in-depth by granting containment overpressure credit whenever an Applicant asked for such credit. Although we may not have seen all the related Committee documentation, we do not believe that ACRS has thoroughly reviewed and recommended this major policy change.

2. Considerable uncertainty continues to exists with regard to determination of BWR pump NPSH adequacy. These uncertainties include non-conservative assumptions in NPSH calculations, reliance on testing which may not conservatively reflect actual conditions, uncertainty and lack of margin in NPSH-required determinations, and non-consideration of chemical effects. Because of the magnitude, importance and significance of these uncertainties we believe that overpressure should be retained as a safety margin rather than used as a credit for NPSH adequacy[3].

3. We agree with the appropriate principle of extended power uprate - using certain safety margins, established many years ago, that are now understood to be excessive through experience or more exact calculations, to permit higher power levels. However, Vermont Yankee's overpressure credit request does not conform to this principle. Vermont Yankee's extended power uprate proposal uses all the available safety margin in NPSH, and then inappropriately seeks to change its design basis to use a properly reserved safety margin in order to achieve the increased power level.

4. For power uprate applications, the staff is not following its own guidance in Regulatory Guide 1.82, Rev. 3[4]. The guidance allows containment overpressure credit when necessary and when the design cannot be practicably altered. However, it appears the

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[3] We are aware of the ACRS interest in NRC staff independent verification of licensee assessments. Because of the importance of this NPSH issue, we asked the NRC staff by letter of June 9, 2004 to independently assess Vermont Yankee's overpressure credit request for LOCA, SBO, ATWS and Appendix R fire events. While the staff has not responded to our letter, it appears from RAI's that the staff is independently verifying the LOCA calculations.

[4] Vice Chairman Wallis queried the staff on this point during the Thermal Hydraulics Subcommittee consideration of Duane Arnold's extended power uprate, September 2001 - See Attachment D to this letter. We are not aware of additional ACRS attention to this matter.

staff now grants containment overpressure credit whenever asked. For Vermont Yankee, containment overpressure credit is not necessary, because the power uprate is not necessary. Also, the Vermont Yankee's design can be practicably altered such that NPSH requirements are met without the need for overpressure credit[5].

It appears the NRC staff is indiscriminately granting containment overpressure credit for extended power uprate, contrary to the policy guidance regarding *need* and *practicable alteration* that are in the Regulatory Guide 1.82, Rev. 3 recommended by the Committee. This appears to be a significant deviation from the Committee's 1997 concurrence that containment overpressure credit be granted *selectively, in a few cases.* Vermont therefore considers this major policy change to be ripe for review and requests the following:

- If the Committee has provided recommendations on the use of containment overpressure credit different from the recommendation in its December 12, 1997 letter quoted above, and different from the guidance in Regulatory Guide 1.82, Rev. 2, we would appreciate it if the documentation of those recommendations would be identified for us.

- We request that the major policy change of granting overpressure credit, described above, be considered by the Committee in its deliberations regarding Vermont Yankee's extended power uprate. We believe the following questions should be considered in reviewing the issue:

  - Should the defense-in-depth provided by unlinked fission product barriers (the containment function and core cooling function) be surrendered (by linking the core cooling function to containment integrity) when there is no need to do so[6] and when the design can be practicably altered[7] to avoid this linkage?

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[5] The backfit rule, 10 C.F.R. §50.109, does not apply because the request for extended power uprate is a voluntary change.

[6] There is no need to change the overpressure credit design basis because it is not necessary for Vermont Yankee to increase its power level.

[7] We believe it is practicable for Vermont Yankee to modify its emergency core cooling system and containment cooling system such that overpressure credit is not required. A possible modification would be to install pumps with different NPSH requirements.

- Is there sufficient continued uncertainty[8] in the sum of:

  1) event calculations that develop the heat loading for torus water,
  2) calculations determining debris loading and strainer head loss,
  3) test results that may not conservatively reflect actual conditions,
  4) uncertainty in NPSH-required values, and
  5) uncertainty resulting from the adverse trend in as-found leakage in Vermont Yankee containment isolation valve leakage rate tests;

  such that the margin provided by containment overpressure should not be surrendered when there is no need to do so and when the design can be practicably altered to avoid this crediting?

- Is risk evaluation methodology sufficiently developed to account for uncertainties in granting overpressure credit, including those identified above, and are the results sufficiently certain, accurate, and reliable to justify using the margin provided by containment overpressure when there is no need to do so and when the design can be practicably altered to avoid this usage?

- Does the post-event, human factors confusion for operators, who would now have to both retain and reduce containment pressure after thirty-two years of training to reduce pressure, merit the containment pressure reliance when there is no need to do so and when the design can be practicably altered to avoid this confusion?

- If this major policy change regarding containment overpressure credit is recommended by the Committee, and if Vermont Yankee's change in design basis is considered, should the precedence created in Section 5.1.4 of Regulatory Guide 1.183[9], regarding the application of current licensing

[8] The calculation uncertainty referred to here includes uncertainties in the methodology accepted by the staff and Committee for such calculations, in the application of this methodology by the Applicant, and in the assumptions and initial conditions chosen by the Applicant.

[9] Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, July 2000:

5.1.4 Applicability of Prior Licensing Basis - The NRC staff considers the

standards, be extended to the grant of extended power uprate, design basis changes for overpressure credit? This method establishes a precedence for evaluating voluntary, major changes of design bases according to current licensing standards.

- If the major policy change is recommended by the Committee that containment overpressure credit should be granted regardless of need or ability to alter design, should there be limits on the percentage or amount of overpressure credited? Is risk methodology sufficiently developed to allow a risk-informed decision on the percentage or amount of overpressure credit to allow? Does the Vermont Yankee request for design basis change fit within these limits?

- We request that Vermont be allowed to present on these issues before the Sub- and Full-Committees, along with the Applicant and staff, in the Committees' deliberations for Vermont Yankee extended power uprate.

- We request that one or both of the Sub- and Full-Committee meetings regarding Vermont Yankee extended power uprate be held in the vicinity of the nuclear plant.

We are aware the staff has granted containment overpressure credit for extended power uprate to other Applicants, and therefore will be reluctant to retreat from this policy. However, Vermont does not believe the implications of this major policy change, as represented by the questions above, have been fully considered. We would greatly appreciate if ACRS

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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.

considerations for the Vermont Yankee extended power uprate could document the Committee's recommendations regarding these questions.

We appreciate your consideration of our request. Please call me if you have questions.

Sincerely,

David O'Brien, Commissioner
NRC State Liaison Officer for Vermont

ATTACHMENT A

State of Vermont Technical Contentions - August 30, 2004
Vermont Yankee Extended Power Uprate

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[1] 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 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

[1] 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.

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 <u>single</u> 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.

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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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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
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Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

Vermont Yankee Nuclear Power Station

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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.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING

DUANE ARNOLD ENERGY CENTER POWER UPRATE REQUEST

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THURSDAY

SEPTEMBER 27, 2001

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ROCKVILLE, MARYLAND

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The ACRS Thermal Phenomena Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 1:00 p.m., Dr. Graham Wallis, Chairman, presiding.

COMMITTEE MEMBERS PRESENT:

DR. GRAHAM WALLIS, Chairman

DR. F. PETER FORD, Member

DR. THOMAS S. KRESS, Member

DR. DANA POWERS, ACRS Cognizant Member

DR. VIRGIL SCHROCK, ACRS Consultant

ACRS STAFF PRESENT:

PAUL A. BOEHNERT, ACRS Staff Engineer

DR. POWERS: Well, I understand what you did.

MS. MOZAFARI: Okay. Then we will just move on. There were open issues that were indicated in the draft safety evaluation. One of them had to do with the start up testing, and the start up testing issue, Mohammed Shuaibi has been following it pretty closely.

The staff has not come to closure on that yet, but it doesn't -- it is not an issue at Duane Arnold at this point, and it will be handled in a license condition when they get to the point where they would trigger the requirement to do start up testing.

And by that time we would have made a decision on the start up testing with our staff. It doesn't become an issue for Duane Arnold at this point. So this will remain an issue that will be addressed when Duane Arnold gets to the power level start up testing where needed.

And then the other issue had to do with MPSH, and Kerry Kavanaugh is going to -- we have a one page handout for that to pass around.

MS. KAVANAUGH: I am Kerry Kavanaugh of the staff. As you heard yesterday, it is the

licensee's position that their licensing basis for the use of containment overpressure is based on margin, which is 2.7 psig.

And they also stated that when they were originally licensed that they were licensed with credit for containment overpressure. The staff agrees that they were licensed for use of containment overpressure from their original licensing basis.

However, the staff does not agree that their licensing basis is based on margin. The staff believes that their licensing basis is based on the magnitude of the overpressure required and the duration of that overpressure as it is required.

This was reflected in their original response to the staff questions on their MPSH when they were licensed. It was in -- their response was a graph that presented the containment pressure versus the time, which represented where the pressure was in the containment over the accident analysis, along with the MPSH requirements during that same time period.

This graph was in the Duane Arnold FSAR and updated FSAR, up until 2000 when it was changed, the figure was changed. During the years, we believe that that graph was the basis for their licensing basis.

When we got to this issue, we had quite a few discussions on it. The staff has reviewed in some respects their MPSH calculations, and we agree with their MPSH analysis for the extended power uprate.

We have sent them a letter, dated September 25th, that basically tells them that any change that increases the magnitude or the duration of the required overpressure than what they are using for their extended power uprate would trigger 10 CFR 50.59 criteria, and would require staff review and approval. That will close the open issue.

DR. POWERS: I guess I understand the approach. Are you telling me that this is an issue that will be resolved if I just wait long enough?

MS. KAVANAUGH: Well, unfortunately, we couldn't resolve it. So they removed that figure from the graph that we were using as their licensing basis. It is now a containment pressure versus suppression pool temperature, which shows that as the pool temperature goes up that they will require containment overpressure.

It doesn't tell you how long they are going to need it, nor does it tell you how much per se, because you really don't know how long they are going to be there.

When we discussed containment overpressures issues with the ACRS staff 3 or 4 years ago, we gave you our approach to resolving the increasing number of licensees that were coming in needing it, and it was based on this time and duration, and an understanding of how much they needed.

And we have not had problems with Duane Arnold in the past because we had this information on the docket. We don't have that now.

CHAIRMAN WALLIS: What is the criteria for acceptability for this time and duration? They mentioned 2.7 psi required, and they showed us that they had much more than that. They didn't say much about time.

MS. KAVANAUGH: They didn't say anything about time.

CHAIRMAN WALLIS: Is time the problem then?

MS. KAVANAUGH: It is very plant specific as to what the criteria is. We have a safety guide, Safety Guide 1, that says that you should not be granting any containment overpressure for your break LOCA analysis.

However, there is a handful of plants with

specifically boilers that cannot meet this requirement, and they were licensed not meeting the safety guide originally and we were aware of this.

As time has gone on, there has been changes with the plants, and most specifically with the BWRs with the strainer issue, and all the BWRs have replaced their ECCS strainers.

And that has changed their headlocks calculations, which is has changed their reliance on containment overpressure, along with other modifications to the plant.

When plants come in needing credit for overpressure, the approach that we have used is that we give them what they need, because we haven't found any licensees willing to change their pumps out of their plants.

So our only opportunity is to evaluate their license, approve their analysis, but give them what they need and allow some room such that they can have some flexibility for operational changes.

Some plants need higher amounts of overpressure and some don't. For Duane Arnold, because they are going up to 209 degrees, I believe is your peak pool temperature, they are going to need approximately 5.8 psig, and I don't remember for what

the time period was, versus two before the EPU.

If you look at another plant with higher pump requirements, they would be needing a higher amount for a lot longer amount of time.

CHAIRMAN WALLIS: Well, I don't quite understand your philosophy of giving them what they need. How is this related to public safety?

MS. KAVANAUGH: Well, since we know what their analysis is, and we are looking at the risk associated and the frequency of having a large break LOCA, we know what their analysis is.

And the analysis for the containment analysis is generally very conservative. They use the super HEX code. They use the ANS 5.1 decay heat, along with a two sigma margin.

Their analysis is done for worst case. So it is generally a very conservative analysis. There really isn't any other way to -- besides changing out the pumps, which would be very expensive for them, to have them meet this safety guide. I mean, the --

CHAIRMAN WALLIS: Well, should I feel good about that? It looks as if you -- that when they need something, you give it to them, but I don't understand the criteria for ever turning them down.

MS. KAVANAUGH: Well, I don't believe

there has been a criteria for turning them down.

CHAIRMAN WALLIS: Well, you might as well just say we have got a rubber stamp here.

MS. KAVANAUGH: What we do is with a lot of care and consideration. I understand your concern, and it has been a hard spot for all of us, but --

CHAIRMAN WALLIS: Is this another case where the rationale is fuzzy?

MR. SHUAIBI: This is Mohammed Shuaibi again. We do go back and look at what is available. It's not that we will give them whatever they want. We will go back and look at what is available and make sure that it is available.

We will look at their containment pressure calculations as we did in this case. So there is margin there. It is not that we will give them what they want, and given a situation where their pumps aren't going to be able to perform.

MS. KAVANAUGH: I mean, the key assumption is that the containment pressure will be there as long as you don't lose that containment pressure. The concern is if that containment pressure isn't there.

CHAIRMAN WALLIS: Well, isn't there then perhaps a power uprate level where you would stop giving them what they need? If they wanted a 25

percent power uprate, and then this would give you a suppression pool temperature of 215 or something -- I mean, there must be some point where you say you can't have what you need.

MS. KAVANAUGH: Well, we haven't reached that evidently yet.

CHAIRMAN WALLIS: Apparently not. How do you know when you reach it?

DR. KRESS: And where do you decide it will be?

MS. KAVANAUGH: No, there is no definition as to where it would be.

CHAIRMAN WALLIS: So there is no speed limit?

MS. KAVANAUGH: But our only control is reviewing the analysis and then getting staff approval. That is our only mechanism for control.

MS. MOZAFARI: Mohammed, do you want to address that?

MR. SHUAIBI: I think clearly that there is a speed limit. I think what your containment is able to withstand is a speed limit, although that is the extreme.

CHAIRMAN WALLIS: There is no speed limit for MPSH per se then?

MR. RUBIN: This is Mark Rubin again, and I will just jump in because I think Mr. Hannon has already left this meeting. Clearly, I would only point out that the safety guide is a not a regulatory requirement.

It is a review guideline, and a very old one additionally. I think perhaps what we are being told is that the staff's evaluation of the plant specific containment analysis is showing that the actual pressure that a good analysis shows is well lin excess of the extra delta-P that they need for the MPSH requirements.

And the staff has confidence that the ECCS systems will successfully operate because of that analytical result, and that public safety is ensured because of that.

DR. POWERS: How does that square with the single failure requirements for the pumps.

MS. KAVANAUGH: I'm sorry?

DR. POWERS: How does that square with the single failure criteria for the pumps?

MS. KAVANAUGH: Well, most plants are not licensed to assume a failure of containment along with a LOCA. I mean, that is beyond their design basis.

MR. RUBIN: If you mean a single failure,

or a single active component failure that would result in increased head requirements, I'm sure that is in the analysis.

MS. KAVANAUGH: Oh, yes, that is in the analysis.

DR. POWERS: All right. But your answer is the one that I was looking for.

MS. KAVANAUGH: Okay.

DR. POWERS: She got it right. She knew what I was talking about, even if I didn't.

CHAIRMAN WALLIS: I guess I would be more reassured if instead of what I heard was give them what they need, if there were some kind of an explanation like it affords here where you have got some kind of prediction that they are making, and this is what they need.

And then you can explain why it is acceptable to be in the region in which they propose to be based on some argument which is quantitative and logical.

MS. KAVANAUGH: Well, I mean, I understand your concern that they do do a containment analysis. It is a minimum containment analysis.

And they use that as a basis to show now much containment pressure they have available. They

don't use all that containment pressure.

CHAIRMAN WALLIS: Well, they believe that the pumps will operate?

MS. KAVANAUGH: They believe that the pumps will operate.

CHAIRMAN WALLIS: And what is your basis for believing the pumps will operate?

MR. SHUAIBI: I think in this case -- and this is Mohammed Shuaibi again -- that we did confirmatory analysis in this case --

MS. KAVANAUGH: For the containment.

MR. SHUAIBI: -- confirmatory containment analysis for this case, and we are comfortable with their values on the pressure that is involved in containment for the scenarios. Unfortunately, we don't have the lead reviewer for that here, and that is what we offered earlier, that he could comment to the full committee and talk about those independent analyses that we did.

DR. POWERS: From a historical point of view, let me see if my understanding -- and you can feel free to correct me if my historical perception in this area is inaccurate.

When we originally licensed these plants, credit was given for overpressure for MPSH because of

the physical fact that it was running and intact, and the coolant loses its density because of its elevated temperature if there was going to be containment overpressure.

That in recent years, we became less confident in that as a safety margin, and we questioned whether that overpressure was appropriate to grant overpressure.

And there are some plants that are licensed to use the containment overpressure. That is an irreversibly fact of life, but we are nervous when we grant these things.

MS. KAVANAUGH: We are getting nervous because they are requiring more. If you look at the original analyses, it was a pound here, and less than a pound. Now we are getting into time periods where they are needing 5 or 6 pounds for several hours.

MS. KAVANAUGH: So, yes, that is where the level of uncomfortable comes from.

DR. SCHROCK: What is the basis of the confirmatory containment analysis? What method is used?

MS. KAVANAUGH: I did not do that analysis. That is something that we can discuss tomorrow, but I believe they used the contain program.

DR. SCHROCK: I am not going to be here tomorrow.

CHAIRMAN WALLIS: There is no tomorrow.

MS. KAVANAUGH: Oh, okay.

MR. SHUAIBI: Again, the lead reviewer on this is not here, but we can discuss that at the full committee meeting. We offered to do that.

DR. SCHROCK: I won't be there either.

MS. KAVANAUGH: But I believe they used the contain program as -- do you remember? You're no help -- the confirmatory analysis code.

MR. BROWNING: This is Tony Browning from Duane Arnold again. The staff was using the contain code, and requested a great deal of data from us so he could benchmark his model to our containment design and specific parameters so that he could do the confirmatory analysis. So that is how it was performed.

DR. POWERS: Any other questions?

CHAIRMAN WALLIS: Well, if these plants don't meet the guidelines, maybe what you need is a new set of guidelines which logically explain a change in position, and explain the rationale for giving credit for these overpressures.

MS. KAVANAUGH: That is a good point.

CHAIRMAN WALLIS: And then set some limits to what is acceptable based on some criterion, which might even be related to risk or something that we can grasp a hold of.

Would it be unreasonable that you recommend that you rewrite the guideline to be more specific, and explicit, and rational?

MS. KAVANAUGH: I believe at one point -- and I don't remember specifically, but I believe it is Reg Guide 182, that also deals with MPSH analysis.

And there was an effort at one time to combine the safety guide in with that, because that deals with vortexing and all kinds of fun stuff, and into one reg guide which would explain that. But I don't know where the staff's effort is on that initiative or not.

MR. BOEHNERT: How many plants are affected by this?

MS. KAVANAUGH: I would say we have 2 or 3 PWRs, which are multiple unit sites; and I would say about 12 BWR sites. You will find that the newer units don't run into this problem. Their MPSH requirements on their pumps are extremely low.

DR. POWERS: Thank you.

MS. MOZAFARI: By way of concluding, I