From: Rick Ennis
To: Brian Sheron
Date: 5/17/04 3:38PM
Subject: Latest letter to Bill Sherman - NPSH

NRR

Brian,

Attached for your information is the latest letter to Bill Sherman on the VY EPU amendment (mostly concerning NPSH). I've also attached a redline/strikeout to show the changes since the version I emailed you on 5/4.

The major changes are :

1) NRC response to question 2.a.1 revised to address guidance document changes we are considering; and

2) NRC response to question 2.f revised to address the safety implications of the proposed change, including a discussion on asking the licensee for risk information.

I plan to continue routing the letter for concurrence. Let me know if you have any comments or would like to discuss before you get for formal concurrence (letter is due out by 6/4).

Thanks,

Rick
415-1420

CC: Allen Howe; Cornelius Holden; Donna Skay; Eric Leeds; Richard Lobel; Robert Dennig; Tad Marsh

B/14

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Mr. William K. Sherman
Vermont Department of Public Service
112 State Street
Drawer 20
Montpelier, VT 05620-2601

Dear Mr. Sherman:

I am responding to your letter dated December 8, 2003, to the U.S. Nuclear Regulatory Commission (NRC), which provided questions regarding the license amendment request from Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. for the Vermont Yankee Nuclear Power Station (VYNPS) dated September 10, 2003. The proposed license amendment would allow an increase in the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWT) to 1912 MWT.

The NRC staff's response to your questions is enclosed. As you are aware, the NRC staff is in the early stages of the review of the VYNPS power uprate amendment request. 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, along with the ongoing NRC inspection program, provide assurance that any issues that could affect safe operation of the plant, related to the proposed power uprate, will be identified. The NRC staff takes its public health and safety mission seriously. As I mentioned during the public meeting on March 31, 2004, the NRC will not approve the proposed amendment unless we are satisfied that safety will be assured.

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,

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

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Vermont Department of Public Service
112 State Street
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Montpelier, VT 05620-2601

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Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271
Enclosure: As stated
cc w/encl: See next page

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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), and March 4, 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). 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 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 during this time period is a key consideration in determining whether the credited pressure will be available.

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. 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. Therefore the NRC staff is considering withdrawing RG 1.1 since the associated guidance has now been replaced by RG 1.82, Revision 3. In addition, the staff is considering updating SRP 6.2.2 to change references to RG 1.82, Revision 1, to RG 1.82, Revision 3, and to delete references to RG 1.1.

Attachment 1 is provided to 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.

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

As discussed in DG 1107, 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. The NRC staff has not considered that the intent of Regulatory Position C.2.1.1.2 was 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.

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. As discussed in Attachment 1, an increase in licensed power level is not the context for crediting containment accident pressure in all cases. Note, the NRC staff does not use any consideration of necessity in determining whether approval of requests for containment accident pressure credit is warranted. 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.

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. In addition, as discussed in Attachment 1, the containment integrity is continuously monitored in the control room.

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 staff intends to request the licensee to provide additional information to further justify relying on containment accident pressure for ECCS pump NPSH. The request will include 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 (formerly DG 1107) does not contain any 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 anticipates requesting 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 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. Licensees' reasons for crediting containment accident pressure included: calculations that incorrectly omitted an important effect (such as underestimating flow losses), an increase in estimated debris loading on BWR ECCS suction strainers in response to NRC Bulletin 96-03, or an increase in suppression pool temperature (due to degradation of the heat transfer capability of the heat exchangers in the suppression pool cooling system). The NRC reviewed all responses to GL 97-04. In some cases, especially those in which credit was taken for containment accident pressure, the NRC performed detailed reviews. The NRC staff formulated and applied acceptance criteria for these reviews. These criteria were not documented in a publically available source at that time. In order to document these criteria for future use, and to make them available to stakeholders, the NRC staff included them in Draft Regulatory Guide (DG) 1107 (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.

The NRC staff briefed the Advisory Committee on Reactor Safeguards (ACRS) twice on the calculation of NPSH and credit for containment accident pressure. The last briefing, on December 3, 1997, discussed the staff's position on credit for containment accident pressure in determining available NPSH during beyond design-basis accidents (DBAs). In a letter to the NRC Chairman, Shirley Ann Jackson, dated December 12, 1997 (ADAMS Accession No. 9712300132), the ACRS concurred with the NRC staff position that selectively granting credit for small amounts of overpressure for a few cases may be justified.

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. 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.

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 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, no debris would be generated and, therefore, the flow losses are considerably lower for these events than for the design-basis LOCA and the available NPSH consequently greater. For these events, containment accident pressure may not be 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.

In summary, consistent with the guidance in RG 1.82, 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. Ensuring containment integrity during this time period is a key consideration in determining whether the credited pressure will be available.

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
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Mr. William K. Sherman
Vermont Department of Public Service
112 State Street
Drawer 20
Montpelier, VT 05620-2601

Dear Mr. Sherman:

I am responding to your letter dated December 8, 2003, to the U.S. Nuclear Regulatory Commission (NRC), which provided questions regarding the license amendment request from Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. for the Vermont Yankee Nuclear Power Station (VYNPS) dated September 10, 2003. The proposed license amendment would allow an increase in the maximum authorized power level for VYNPS from 1593 megawatts thermal (MWT) to 1912 MWT.

The NRC staff's response to your questions is enclosed. As you are aware, the NRC staff is in the early stages of the review of the VYNPS power uprate amendment request. 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, along with the ongoing NRC inspection program, provide assurance that any issues that could affect safe operation of the plant, related to the proposed power uprate, will be identified. The NRC staff takes its public health and safety mission seriously. As I mentioned during the public meeting on March 31, 2004, the NRC will not approve the proposed amendment unless we are satisfied that safety will be assured.

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,

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: As stated

cc w/encl: See next page

Mr. William K. Sherman
Vermont Department of Public Service
112 State Street
Drawer 20
Montpelier, VT 05620-2601

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Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271
Enclosure: As stated
cc w/encl: See next page

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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), and March 4, 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). 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 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 during this time period is a key consideration in determining whether the credited pressure will be available.

The NRC's ~~position~~guidance regarding whether it is acceptable to credit containment accident pressure has evolved over the years. ~~In order to understand~~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. 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. Therefore the NRC staff is considering withdrawing RG 1.1 since the associated guidance has now been replaced by RG 1.82, Revision 3. In addition, the staff is considering updating SRP 6.2.2 to change references to RG 1.82, Revision 1, to RG 1.82, Revision 3, and to delete references to RG 1.1.

Attachment 1 is provided to in order to explain the NRC's current position~~, Attachment 1 provides further details on this issue.~~ 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.

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

As discussed in DG 1107, ~~Regulatory Guides (~~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.

<u>DPS Questions 2.b and 2.c</u>

(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?

<u>NRC Response to DPS Questions 2.b and 2.c</u>

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). RG's 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. The NRC staff has not considered that the intent of Regulatory Position C.2.1.1.2 was 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.

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. As discussed in ~~Enclosure~~Attachment 1, an increase in licensed power level is not the context for crediting containment accident pressure in all cases. Note, the NRC staff does not use any consideration of necessity in determining whether approval of requests for containment accident pressure credit is warranted. 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.

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. In addition, as discussed in Attachment 1, the containment integrity is continuously monitored in the control room. ~~Additional containment pressure testing is not considered to be required if the maximum TS-allowable containment leakage is assumed in the calculation of the available NPSH. This assumption will be verified as part of the staff's review of the proposed request.~~

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

~~In an NRC Staff Requirements Memorandum (SRM) dated August 25, 1997, the Commission provided a discussion on safety and compliance. The SRM stated that an activity is deemed to be safe if the perceived risks are judged to be acceptable. The Atomic Energy Act of 1954, as amended, established "adequate protection" as the standard of safety on which NRC regulation is based. Adequate protection is presumptively assured by compliance with NRC requirements. The SRM stated that compliance simply means meeting applicable regulatory requirements. As is the case for any proposed license amendment, the NRC staff review determines if the proposed changes would be in compliance with the applicable regulatory requirements. Therefore, if the NRC review determines that the proposed changes would be in compliance with the applicable regulatory requirements, there is reasonable assurance that the proposed change is safe. The basis for allowing credit for containment accident pressure is~~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 staff intends to request the licensee to provide additional information to further justify relying on containment accident pressure for ECCS pump NPSH. The request will include 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 (formerly DG 1107) does not contain any 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~~[extended power uprate]~~, 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?

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 anticipates requesting 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 ~~in this area~~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~~ 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 ~~U.S. Nuclear Regulatory Commission (~~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 ~~Section~~ 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 ~~(psig)~~ 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. Licensees' reasons for crediting containment accident pressure included: calculations that incorrectly omitted an important effect (such as underestimating flow losses), an increase in estimated debris loading on BWR ECCS suction strainers in response to NRC Bulletin 96-03, or an increase in suppression pool temperature (due to degradation of the heat transfer capability of the heat exchangers in the suppression pool cooling system). The NRC reviewed all responses to GL 97-04. In some cases, especially those in which credit was taken for containment accident pressure, the NRC performed detailed reviews. The NRC staff formulated and applied acceptance criteria for these reviews. These criteria were not documented in a publically available source at that time. In order to document these criteria for future use, and to make them available to stakeholders, the NRC staff included them in Draft Regulatory Guide (DG) 1107 (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.

The NRC staff briefed the Advisory Committee on Reactor Safeguards (ACRS) twice on the calculation of NPSH and credit for containment accident pressure. The last briefing, on December 3, 1997, discussed the staff's position on credit for containment accident pressure in determining available NPSH during beyond design-basis accidents (DBAs). In a letter to the NRC Chairman, Shirley Ann Jackson, dated December 12, 1997 (ADAMS Accession No. ~~9712300132~~ 9712300132), the ACRS concurred with the NRC staff position that selectively granting credit for small amounts of overpressure for a few cases may be justified.

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. 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.

The present VYNPS ECCS and containment licensing basis, as with all licensed nuclear power plants, is derived from ~~design-basis accident (~~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 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, no debris would be generated and, therefore, the flow losses are considerably lower for these events than for the design-basis LOCA and the available NPSH consequently greater. ~~C~~For these events, containment accident pressure may not be need to be credited. For cases where it is credited, the amount of containment accident pressure is ~~sometimes~~typically less than that credited for ~~these events, although the design-basis LOCA conditions would be bounding. Containment accident pressure is not credited for shutdown conditions.~~
the postulated LOCA.

In summary, consistent with the guidance in RG 1.82, 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. Ensuring containment integrity during this time period is a key consideration in determining whether the credited pressure will be available.

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.