Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Vermont Yankee
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500
Tel 802 257 5271

October 21, 2005

Docket No. 50-271
BVY 05-098
TAC No. MC0761

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: **Vermont Yankee Nuclear Power Station**
**Technical Specification Proposed Change No. 263 – Supplement No. 38**
**Extended Power Uprate – Containment Overpressure Credit**

Reference:

1) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate," BVY 03-80, September 10, 2003

2) U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Rev. 1, November 2002

3) U.S. Nuclear Regulatory Commission, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Regulatory Guide 1.82, Rev. 3, November 2003

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

As part of the extended power uprate (EPU) submittal for VYNPS (Reference 1), Entergy proposed taking credit for containment accident pressure to provide adequate net positive suction head (NPSH) to the emergency core cooling system (ECCS) pumps. Section 4.2.6 of the Power Uprate Safety Analysis Report (PUSAR) (Attachments 4 (proprietary) and 6 (non-proprietary) of Reference 1) provides the evaluation of NPSH for the ECCS pumps following a design basis loss-of-coolant accident. That evaluation was performed using deterministic methods and incorporated conservative design and operational assumptions regarding containment response and pump performance. The evaluations and analyses presented in the PUSAR are based on the use of traditional engineering methods that include worst-case assumptions and other engineering conservatisms.

A001

To provide a supplementary assessment of the acceptability of crediting containment accident pressure and the associated risk impacts of EPU on functional and system-level success criteria including ECCS pump cavitation, Entergy is supplementing previous EPU analyses with the risk insights provided herein. The risk evaluations presented in Attachment 1 follow the guidance provided in Regulatory Guide 1.174 (Reference 2), which encourages the use of probabilistic risk assessment techniques to improve safety decisionmaking and improve regulatory efficiency. Those risk evaluations support the analyses provided in the PUSAR regarding crediting containment accident pressure.

The changes in plant operation proposed in the EPU submittal for VYNPS are consistent with currently approved NRC staff positions (specifically, Reference 3). In addition, the evaluations and risk information provided in Attachment 1 further justify the proposed changes because those risk insights complement the deterministic assessment conducted under the traditional defense-in-depth approach. The common conclusion from the analyses, using both approaches, is expected to demonstrate that the proposed crediting of containment accident pressure represents a small change in risk and meets NRC's policy statement for safety goals.

Entergy has not completed the risk analyses addressed herein, specifically the quantitative assessment of risk impact. That information will be submitted to the NRC staff no later than October 28, 2005.

There are no new regulatory commitments contained in this submittal.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 21, 2005.

Sincerely,

Jay K. Thayer

Jay K. Thayer
Site Vice President
Vermont Yankee Nuclear Power Station

Attachments (1)

cc: (see next page)

cc: Mr. Samuel J. Collins (w/o attachment)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O 8 B1
Washington, DC 20555

USNRC Resident Inspector (w/o attachment)
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

BVY 05-098
Docket No. 50-271

# Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 -- Supplement No. 38

Extended Power Uprate - Containment Overpressure Credit

Total number of pages in Attachment 1 (excludina this cover sheet) is 26.

## Executive Summary

To provide an alternative assessment of the acceptability of crediting containment accident pressure and the associated risk impacts of extended power uprate (EPU) operation for the Vermont Yankee Nuclear Power Station (VYNPS), Entergy is supplementing previous EPU analyses with the risk insights provided herein. The risk evaluations presented follow the guidance provided in Regulatory Guide 1.174, which encourages the use of probabilistic risk assessment techniques to improve safety decisionmaking and improve regulatory efficiency. The risk evaluations support the analyses provided in the Power Uprate Safety Analysis Report regarding crediting containment accident pressure.

The changes in plant operation proposed in the EPU submittal for VYNPS are consistent with currently approved NRC staff positions. In addition, the evaluations and risk information provided further justify the proposed changes because those risk insights complement the deterministic assessment conducted under the traditional defense-in-depth approach.

The proposed change crediting containment overpressure (COP) meets current regulations and is consistent with NRC staff positions, including Regulatory Guide 1.82, Rev. 3. The deterministic evaluations and analyses were performed in accordance with regulatory requirements and demonstrate that the level of protection necessary to avoid undue risks will be maintained.

Defense-in-depth principles are maintained, and realistic analyses show there is adequate net positive suction head (NPSH) for emergency core cooling pumps without crediting COP. The proposed change is consistent with the defense in depth philosophy as success measures are incorporated into the design, construction and operations to compensate for potential failures in protection and safety measures. Realistic analyses with single failure assumptions show that adequate protection is afforded to mitigate the release of radioactivity for postulated accidents.

Probabilistic safety analyses (PSA) underscore the many conservatisms in traditional engineering calculations regarding low pressure emergency core cooling system pump NPSH. At the same time, PSAs indicate that containments are robust and highly reliable structures.

The VYNPS PSA model is highly detailed and includes a wide variety of initiating events, modeled systems, operator actions, and level of detail. The model, having undergone several updates, adopts the large event tree / small event tree approach using the support state methodology embodied in the RISKMAN code for quantifying core damage frequency.

The common conclusion from the analyses, using both deterministic and risk-informed approaches, is expected to demonstrate that the proposed crediting of containment accident pressure represents a small change in risk and meets NRC's policy statement for safety goals.

Introduction

The proposed change in VYNPS' licensing basis (LB) for EPU includes crediting containment overpressure (COP) in ensuring adequate NPSH to ECCS pumps following certain postulated events (i.e., the design basis loss-of-coolant accident (LOCA) and anticipated transients without scram (ATWS)). The current licensing basis for VYNPS does not credit the COP that is present in containment following any postulated accident or transient event.

Background

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," specifies five elements for consideration when applying risk-informed licensing basis changes. These five elements consider engineering issues and apply risk insights in the following manner:

1. The proposed change meets the current regulations.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When the proposed changes result in an increase in the core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

This approach supplements prior deterministic analyses with risk-informed evaluations using methodologies previously found acceptable by the NRC staff in other applications although not previously applied to COP. This approach supports the evaluation of design basis LOCA and non-design basis ATWS events for which COP credit is requested in this change to the licensing basis of VYNPS under EPU conditions. Additionally, this approach represents a more realistic treatment of assumptions used in accident analyses.

## Element #1 - Current Regulations

The proposed change in LB crediting COP meets current regulations and is consistent with currently approved NRC staff positions (e.g., Regulatory Guide 1.82). Entergy has analyzed this change in LB using traditional engineering analyses and has documented the results in the Power Uprate Safety Analysis Report (PUSAR) for VYNPS. The information provided herewith supplements the analyses provided in the PUSAR and related supplements to the request for license amendment with risk-informed evaluations to further demonstrate that the level of protection necessary to avoid undue risk to public health and safety (i.e., "adequate protection") will be maintained following approval of the change in LB crediting COP. The change in LB for COP is consistent with the following regulatory staff positions regarding the adequacy of ECCS and containment integrity necessary to maintain sufficient COP for the duration of the time credited:

- 10 CFR 50.54 (o)
- Appendix J to 10 CFR 50
- Regulatory Guide 1.163
- Regulatory Guide 1.82
- 10 CFR 50.46 and Appendix K to 10 CFR 50
- Draft GDC 44, 49, and 52
- 10 CFR 50.62

The NRC staff has allowed limited credit for a containment pressure that is above the vapor pressure of the sump fluid (i.e., an overpressure) to satisfy NPSH requirements on a case-by-case basis. The NRC staff's prior acceptance of COP in the analyses for other plants provides precedence for crediting COP in the VYNPS analyses.

## Element #2 – Defense-in-Depth Attributes

BWR ECCS Pump Designs

Adequate pump NPSH can become an issue in the maintenance of core cooling when suppression pool level is low or pool temperature is high, e.g., during loss of decay heat removal (DHR) accidents where ECCS pumps may be required to pump hot water near saturation temperature.

BWR/4 product lines used ECCS pump designs such as vertical or deep well pumps. These pumps may require greater NPSH for successful operation relative to later pump designs. Industry tests have shown that the vertical pump designs are capable of adequate short term (~24 hrs) operation at NPSH conditions significantly below manufacturer recommended design NPSH (e.g., 60-65% of the design NPSH limit).

Low pressure ECCS pump design NPSH is typically based on manufacturer recommended NPSH limits. The manufacturer recommended NPSH limit includes an operational design margin. The amount of margin depends on the specific pump design and the operating conditions of the pump.

Design Basis LOCA

Crediting COP in determining NPSH margin for the design basis LOCA and ATWS events does not contravene defense-in-depth principles. The VYNPS design basis LOCA analysis for containment applies conservative bounding input assumptions depending on the specific safety margin to be demonstrated. Thus, to determine if adequate margins relative to failure of the containment boundary can be demonstrated, the analysis makes assumptions regarding equipment availability and operator actions that are intended to maximize the calculated containment pressures (i.e., no containment leakage and no containment spray) and temperatures (only one RHR train available for heat removal) for comparison to design limits. For analyses to determine if adequate NPSH margin exists, the containment analysis assumes a bounding value for leakage and operation of containment sprays to minimize containment pressure, while maintaining the same assumption regarding the availability of only one RHR train in order to maximize the calculated suppression pool temperature. These analyses are done in the same manner whether or not COP is credited in the calculations of NPSH margins. In both cases the single failure assumption is applied (i.e., no more than one single active failure) in determining the availability of systems, components, or structures. Furthermore, the application of overpressure credit does not require any additional operator actions. Therefore, the application of COP in the determination of NPSH margin has no impact on the defense-in-depth philosophy.

Although VYNPS' current licensing basis does not assume passive failures of the containment, the single failure criterion can be applied to the containment analysis in the following manner. If it is assumed (none have been identified) that a single failure results in leakage from the containment that sufficiently exceeds a value that would result in pressure being less than that credited for NPSH margin in the DBA-LOCA analysis, then all pumps would still be available. Since the single failure is the containment pressure boundary, both RHR trains can be credited in the analysis; in which case, COP is unnecessary. The response to RAI SPSB-C-10[1] showed that the peak suppression pool temperature.in this case would be 169°F, which is well below the temperature at which any overpressure credit needs to be applied. Thus, the credit applied for overpressure for the DBA-LOCA does not increase the reliance on containment integrity. Similarly, the ability of the ECCS to provide adequate core cooling is not degraded by the application of the overpressure credit.

Since the application of overpressure credit does not involve any changes to systems, structures, or components, there is no impact on system redundancy, independence, and diversity.

The application of containment overpressure will not have an impact on any programmatic requirements or activities (e.g., Appendix J to 10 CFR 50).

The application of COP does not require any changes to or additional operator actions. The only operator actions postulated for the DBA-LOCA are the initiation of containment cooling and throttling core spray (CS) flow. Emergency Operating Procedures contain guidance for assuring adequate NPSH based on pump flow rate, suppression pool temperature, and containment pressure in the form of limit curves. These curves are based on the characteristics of the pumps and are not affected by the crediting of overpressure as a licensing basis. Operators would only throttle CS if actual conditions dictate it in accordance with the limit curves. Thus, the crediting of overpressure in the DBA-LOCA analysis does not place any additional burden on the operators.

<u>ATWS</u>

The ATWS analysis does not apply any additional single failure to the ATWS event since the "failure to scram" scenario already assumes more than a single failure. This is acceptable given the relative low probability of the event. Thus, containment integrity is assumed and will result in adequate overpressure for the RHR pumps. Core spray pumps are not normally assumed to be required for an ATWS event.

The amount of overpressure required (1.2 psi maximum) and the duration for which it is required (about one hour) is relatively small for the ATWS event. Thus, it is relatively insignificant compared to the DBA-LOCA discussed above, and the same conclusions regarding no impact on defense-in-depth apply.

<u>Operator actions</u>

Plant specific emergency operating procedures (EOPs) may have cautions or restrictions on the ECCS pumps associated with NPSH or vortex limits. These procedural restrictions may apply except for cases when RPV injection is needed for adequate core cooling. As such, NPSH curves would not prohibit use of ECCS pumps when they are required for core cooling.

The VYNPS EOPs are consistent with this typical approach to reactor level control. Step RC/L-15 of the VYNPS Reactor Level Control EOP directs the operator to restore reactor water level using all available injection systems and exceeding NPSH limits if necessary. Furthermore, as

[1] Supplement 8 to the Vermont Yankee EPU application, BVY 04-058, Attachment 1, page 149, July 2, 2004

directed in the EOPs, the plant operator will throttle the system to stay within NPSH limits. Therefore, the VYNPS EOP directions are consistent with the crediting of COP for NPSH considerations.

Reasonable Balance

RG 1.174 encourages the use of risk analysis techniques to help ensure and explain how the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among (1) prevention of core damage, (2) prevention of containment failure, and (3) consequence mitigation. Assuming no change in the probability of containment failure due to EPU and the ability to mitigate consequences, the defense-in-depth philosophy is maintained.

Preservation of Multiple Barriers to the Release of Radioactivity

The design and operating practices that compensate for potential failures in protection and safety measures ensure that multiple barriers are adequately maintained to preclude or mitigate the release of radioactivity in the event of a postulated accident. Through a succession of passive barriers extending from the nuclear fuel to its cladding, the reactor coolant pressure boundary and containment structures, effective containment of radioactivity is provided through this defense-in-depth. With the objective of maintaining the integrity of barriers, the proposed change crediting containment accident pressure to ensure adequate NPSH to ECCS pumps was evaluated with the following conclusions:

1. The proposed change does not result in a significant increase in the existing challenges to the integrity of the barriers. Peak cladding temperatures remain well within acceptance limits under 10 CFR 50, Appendix K analyses; and reactor coolant system and primary containment pressures are not significantly increased under EPU accident conditions.

2. The proposed change does not significantly change the failure probability of any individual barrier. Failure probabilities were addressed in the original VYNPS EPU probabilistic safety assessment and evaluated to be acceptably low.

3. The proposed change does not significantly increase the likelihood of a barrier failure compared to existing conditions. The likelihood of failure of the containment barrier is not significantly increased as a result of the proposed change. The introduced dependency of ECCS performance based on the maintenance of containment integrity using traditional engineering analyses assumes conservatisms that are unlikely to simultaneously occur.

4. The overall redundancy and diversity among the barriers are unchanged and remain sufficient to fulfill the following risk acceptance guidelines:

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

b. The proposed change avoids over-reliance on programmatic activities to compensate for weaknesses in plant design.

c. The proposed change adequately preserves system redundancy, independence, and diversity commensurate with the expected frequency of challenges, consequences of failure of the system, and associated uncertainties.

d. The proposed change preserves defenses against potential common cause failures and assess the potential introduction of new common cause failure mechanisms.

e. The proposed change does not degrade the independence of barriers.

f. The proposed change preserves defenses against human error.

g. The proposed change fulfills the intent of the draft General Design Criteria in proposed Appendix A to 10 CFR 50, which is part of the current licensing basis for VYNPS.

## Element #3 – Maintaining Relevant Safety Margins

### Design Basis Accident Licensing Analysis

The NRC staff has selectively granted credit for containment pressure in NPSH determination in design basis calculations.

The issue of design basis assumptions and calculations regarding low pressure ECCS pump required NPSH ($NPSH_R$) should not have substantial influence on the realistic assessment of ECCS pump performance in the PSA. The PSA is a realistic analysis that considers probabilistic factors that impact pump availabilities and capabilities, and does not assume blanket worst case assumptions that are typically inherent in design basis calculations (e.g., 102% reactor power, bounding decay heat curves, fouled heat exchangers).

If the available NPSH is reduced due to high suppression pool temperature, then events such as containment failure, venting, or initiation of containment sprays can result in a rapid decrease in containment pressure. This rapid decrease in containment pressure can cause a decrease in the saturation pressure and result in substantial flashing of saturated fluid to steam, thus, causing steam binding within the pumps or suction piping. Such phenomena are incorporated in PSAs as a potential failure of low pressure ECCS pumps in the appropriate accident sequences.

Available NPSH ($NPSH_A$) can be substantially reduced by clogging of suppression pool suction strainers. This can occur due to debris in the suppression pool that is pre-existing or the result of an event such as a LOCA. This effect on NPSH needs to be included in the realistic assessment of ECCS pump operability under severe accident conditions. The VYNPS ECCS suppression pool strainers were replaced in 1998 with new strainers designed to provide significantly greater surface area. Even if the large amount of drywell debris assumed in the DBA is transported to the suppression pool during a design basis accident, the pressure drop across the new strainers is not significant (approximately half a foot of head loss) and is unchanged for EPU.

### NPSH Under Severe Accident Conditions

Within the assessment of severe accidents, substantial, degraded conditions may exist. These degraded conditions may involve low suppression pool level and high suppression pool temperature. Both conditions lead to situations where available NPSH may fall near or below the recommended design NPSH. However, such conditions do not mean that the pumps cannot perform adequately in the short-term (e.g., the 24 hour mission time of the PSA). In fact, plant tests at Browns Ferry (documented in NUREG/CR-2973, Loss of DHR Sequences at Browns Ferry Unit One) has shown significant NPSH margin exists for ECCS pumps of the vertical pump design characteristic of BWR/4 plants. Both tests used a lowered suppression pool level as the method of reducing the available NPSH. The results show that the vertical design RHR pumps can operate at significantly reduced NPSH compared to the design NPSH. NUREG/CR-2973 states the following:

> *"In-plant testing at Browns Ferry has shown that the RHR pumps can operate down to about 65% of the manufacturer recommended minimum NPSH with the*

*following consequences: 10% degradation of developed pump head, acceptable pump motor vibration, but severe audible cavitation. This would not jeopardize short-term operation although impeller cavitation damage would be expected in the long-term. ... The NPSH margin for acceptable RHR pump operation can be extended by operator action, throttling the flow as necessary to reduce the RHR pump discharge from the rated flow of 0.63 $m^3/s$ (10,000 gpm)."*

These test results and the knowledge of the inherent margin in design NPSH limits and accident calculations are often used in industry PRAs to more realistically model the NPSH limits of low pressure ECCS pumps.

Typical of other industry PRAs, the VYNPS PSA uses this test information to credit more realistic NPSH curves for the low pressure ECCS pumps in some accident sequences. This credit is supported by a VYNPS engineering assessment in which a review of the Browns Ferry tests was performed and compared to the VYNPS design. A pump specialist at GE Nuclear was also consulted as part of the engineering assessment. The review confirmed that the Browns Ferry information is applicable to VYNPS.

VYNPS ECCS Pump Margin Assessment

The PSA model provides realistic criteria required for successful operation of the RHR and CS pumps. One of the criteria is that the pumps will operate at reduced $NPSH_R$ compared with the pump manufacturer's $NPSH_R$ value used in the UFSAR analysis. The basis for the RHR pump has been previously documented and a synopsis is repeated in this evaluation. The basis for the CS pumps, based on vendor information and engineering judgment, is also provided.

The $NPSH_R$ values typically provided by pump manufacturers are based on the NPSH value where pump discharge head decreases by 3%. The pump vendors provide a conservative value to provide operability margin and to ensure that there are no product liability issues. At the 3% head drop value, pumps will operate continuously without physical pump degradation.

VYNPS' RHR and CS pumps were manufactured by Bingham Williamette (now Sulzer Bingham). The RHR pumps are single stage vertical pumps, Model 16 x 18 x 26 CVIC with a single suction entrance. The CS pumps are also single stage vertical pumps but have a double suction entrance, Model 12 x 16 x 14-1/2 CVDS.

A 1998 study conducted by Sulzer Bingham entitled "NPSH Study of RHR & CS pumps," provided time dependent recommended minimum NPSH values for a pump life of 8000 hours with varying NPSH values for operating times from 0 to 7 hours, 7 to 20 to 100 hours, and 100 hours to 8000 hours (Attachments 1 and 2).

The vendor recommended NPSH value for time frame of 0 – 7 hours was used for the short-term phase of the design basis LOCA NPSH analysis and the long term phase conservatively and appropriately used the 100 hour NPSH values.

The minimum $NPSH_A$ values are applicable at any time and are not time sequence dependent, even for the design basis case. For example, during the initial phases of a LOCA, the $NPSH_A$

could be well above the >100 hour value and as the temperature torus heats up to its maximum temperature, the 7 hour duration $NPSH_A$ value would govern.

RHR Pump

NPSH testing was conducted at Browns Ferry on the same model (different size) RHR pump as those installed at VYNPS. These tests showed that the RHR pump operated without significant vibration at roughly 60% to 70% of the vendor recommended $NPSH_R$ values for flow rates near the pump design value. These tests provide the PSA basis for RHR pump success with reduced $NPSH_A$. At these NPSH values pump head can be reduced ~12% but the pump will still be operating above the "knee" of the pump curve.

CS Pump

The Browns Ferry NPSH tests were specific to the RHR pumps. However, there are numerous similarities between the RHR and CS pumps. The Browns Ferry RHR pumps and the VYNPS RHR pumps are the same model. The CS pumps were manufactured by the same manufacturer. All of the pumps are single stage vertical pumps which are inherently rugged. Other than the double suction entrance, the CS pumps are similar in design.

The vendor study provided in Attachment 5 of calculation VYC-0808, Rev. 8[2] has the following additional information on CS pump operation at $NPSH_R$ values less than the recommended value used for design basis analysis:

- Section III.B under CS pumps states: "The most complete and representative test is T-176101-D/G (see also SBPI Curve No. IIc)." Item 3 of the same section states that "Operating for short durations at $NPSH_R$ – 3% to $NPSH_R$ – 6% should not be detrimental to the pump life in this service."

- Curve IIc (Attachment 3) shows that the $NPSH_R$ values with a 6% head reduction are approximately 19.5' at a flow rate of 3000 gpm and 21.5' at a flow rate of 3500 gpm. The 6% head reduction values are a fraction of the vendor recommended values which for 3000 gpm for 0 – 7 hours is 24' (vs. 19.5' which is 81%) and for long term is 29' (vs. 19.5' which is 67%).

- The vendor NPSH information for a flow rate of 3500 gpm was extrapolated to yield a long term value of 29.6' (vs. 21.5' which is 73%).

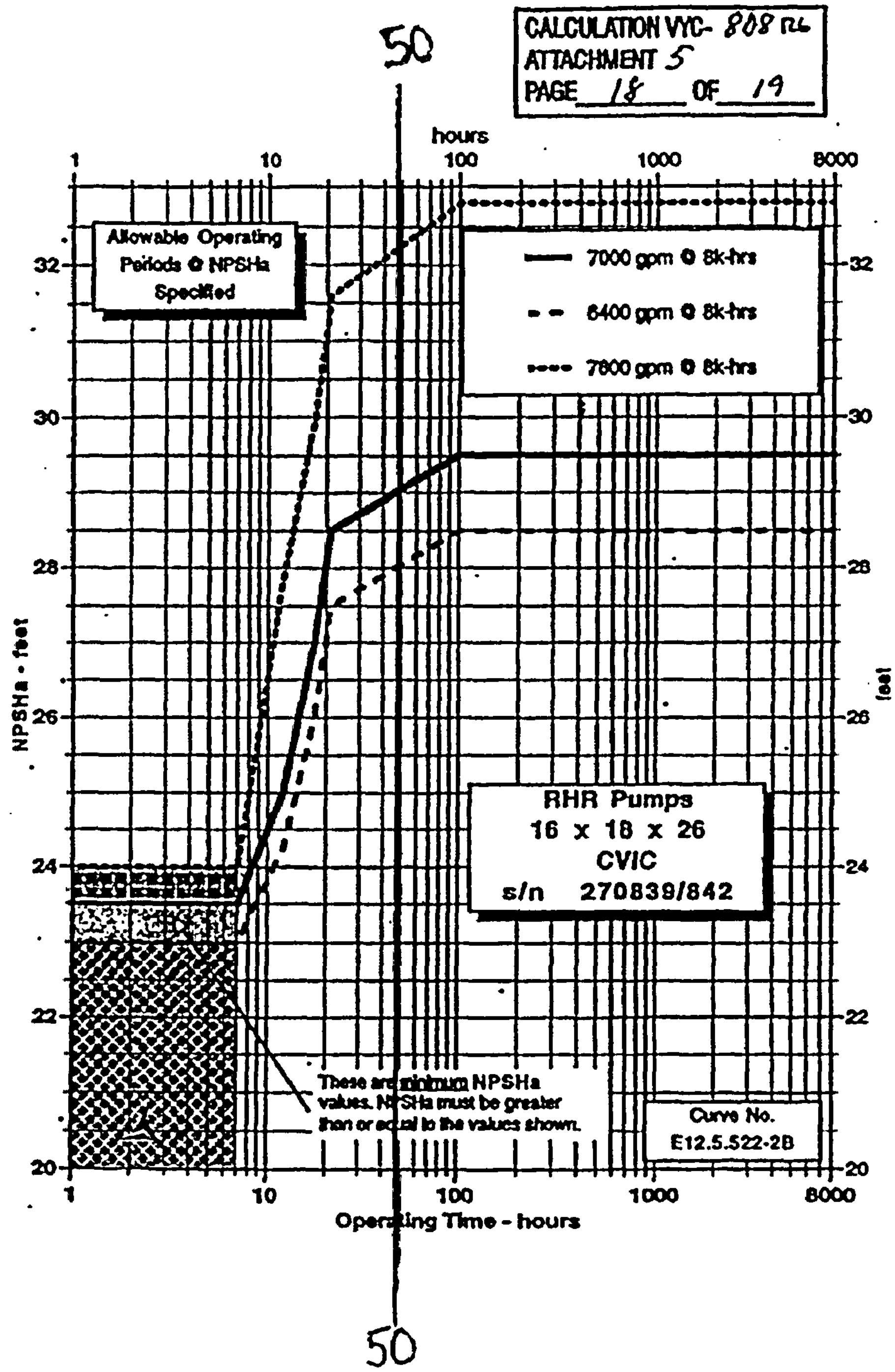
- Curve IIc shows that there is approximately a 3' difference between the $NPSH_R$ values at 3% head reduction and the 6% head reduction value. Based on engineering judgment, it is reasonable to assume that at 9% or 12% head reduction (RHR head reduction at Browns Ferry test) that the $NPSH_R$ would be reduced. It is also our engineering judgment based on the similarity of the RHR and CS pumps and the fact that CS is operated below its design point that the pump will still be operating above the "knee" of the curve.

[2] Calculation VYC-0808, Rev. 8 was submitted to the NRC as part of EPU Supplement 18, Entergy letter BVY 04-106, October 5, 2004

Conclusions

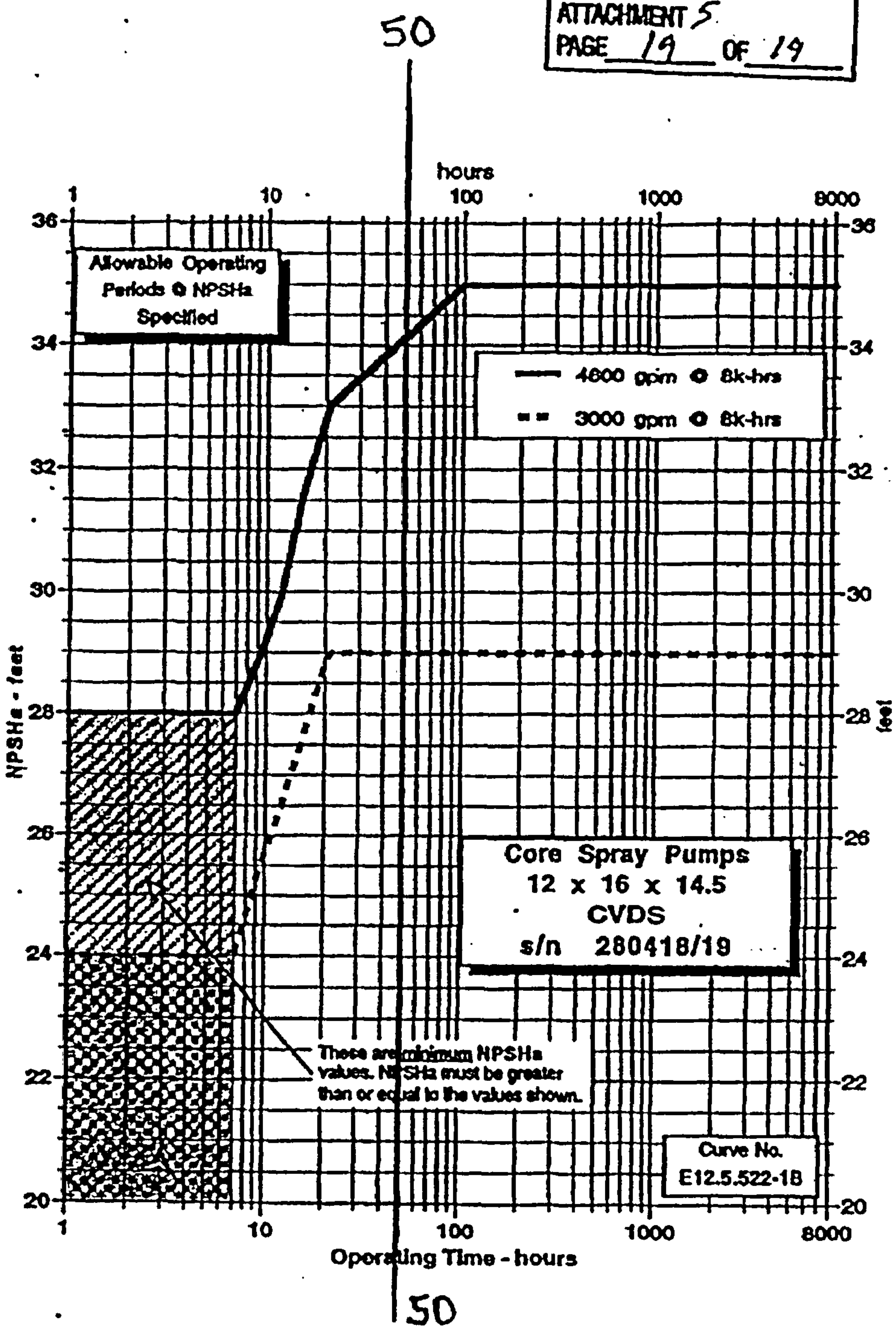
- Pump information from the vendor indicates that the CS pumps can operate at approximately 2/3 of the vendor-recommended design value. With engineering judgment this value can be reduced to approximately 60%. This documents that the PSA success criteria for the CS pumps is valid.
- From a realistic PSA perspective, the RHR and CS pumps could operate without credit for COP and pump life would not be adversely affected.

CALCULATION VYC- 808 R4
ATTACHMENT 5
PAGE 18 OF 19

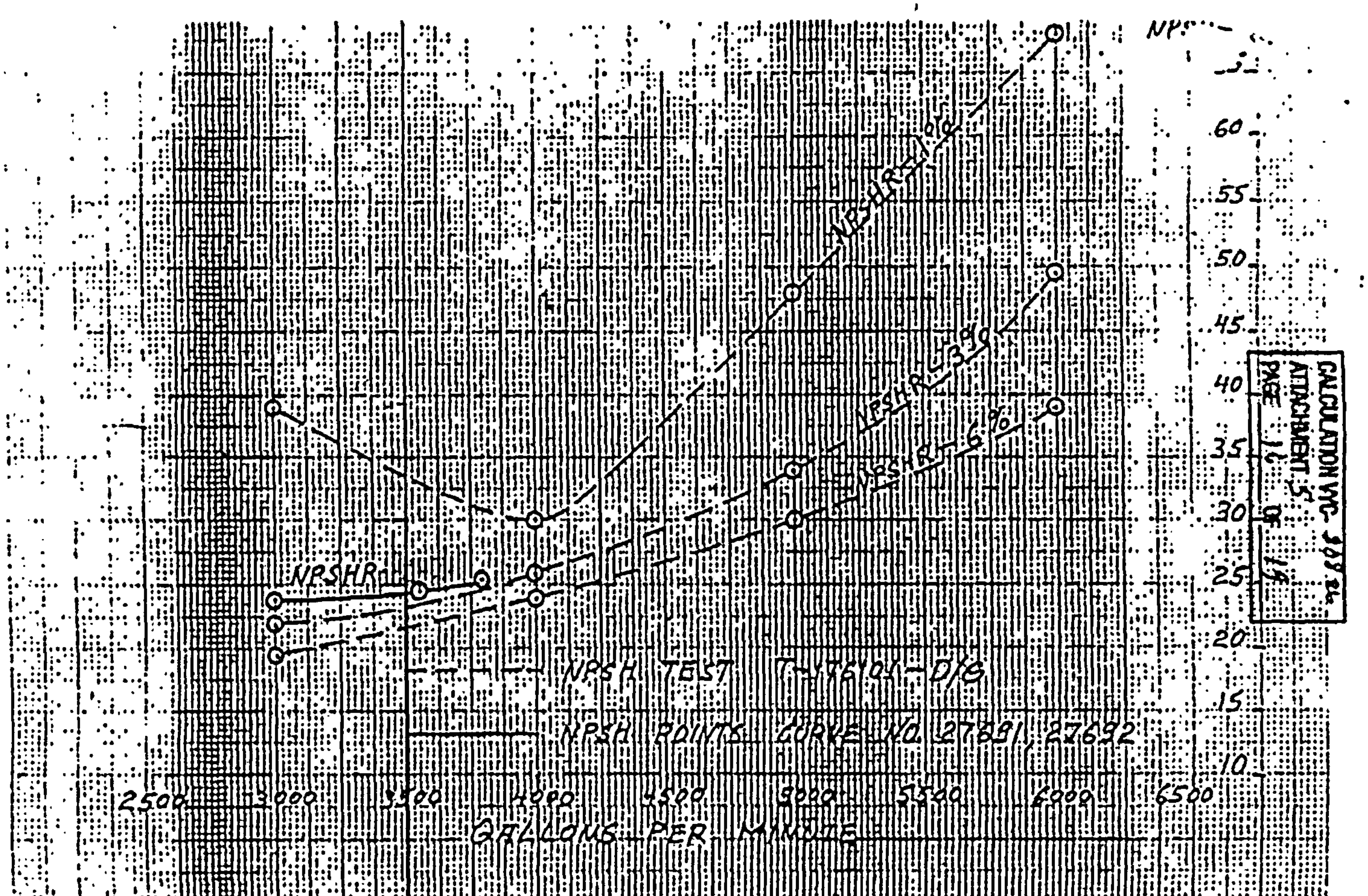



7 30P59VY.DG4 23:45 Tue May 19, 98

CALCULATION VYC- 808 R1
ATTACHMENT 5
PAGE 19 OF 19




4 30P59VY.DG4 23:51 Tue May 19, 98




| | | | | | | |
|---|---|---|---|---|---|---|
| YANKEE ATOMIC ELECTRIC<br>VERMONT YANKEE<br>CS PUMPS<br>S.O. NO. 280.418/419 | PUMP ENGINEERING DEPT.<br>**SULZER BINGHAM PUMPS INC.**<br>18 MAR 98, R.L. | IMPELLER MAX. DIA. 14.500" | 12 × 16 × 14½ CVDS PUMP | | | |
| | | MIN. DIA. | DIA. IMPELLER 12.500" | IMPELLER PATT. 1213 CVDS-1 | 3582 R.P.M. | |
| | | EYE AREA 72.2 SQ. IN. | N.P.S.H.R. REQUIRED | REFERENCE | CURVE NO. II C | |

@ 3500 21.5' 6%
@ 3000 19.1' 6%

Att 3

## Element #4 – Risk Impact Assessment

The purpose of this evaluation is to provide a risk assessment of using containment accident pressure or containment overpressure in evaluating the Net Positive Suction Head (NPSH) requirements for emergency core cooling and containment heat removal pumps. Details of the PSA performed to support to this risk evaluation, including quantitative results, will be subsequently submitted.

### PSA Model Description

The VYNPS PSA model uses widely-accepted PSA techniques for event tree and fault tree analysis. Event trees are constructed to identify core damage and radionuclide release sequences. The event tree "top events" represent systems (and operator actions) that can prevent or mitigate core damage. Fault trees are constructed for each system in order to identify the failure modes. Analyses of component failure rates (including common cause failures) and human error rates are performed to develop the data needed to quantify the fault tree models.

In PSA terminology, the VYNPS PSA modeling approach can be characterized as an "event tree linking" approach. The event tree top events correspond to systems, as opposed to the more general functions used by some analysts. Our approach divides the plant systems into two categories:

1. Front-Line Systems, which directly satisfy critical safety functions (e.g., Core Spray and RHR Torus Cooling), and
2. Support Systems, which are needed to support operation of front-line systems (e.g., ac power and Service Water).

Front-line event trees are linked to the end of the Support System event trees for sequence quantification. This allows us to establish the status of all support systems for each sequence before the front-line systems are evaluated. Quantification of the event tree and fault tree models is performed using the RISKMAN code. The Support System and Front-Line System event trees are "linked" together and solved for the core damage sequences and their frequencies. This calculation is performed with the RISKMAN code's event tree module. Each sequence represents an initiating event and combination of Top Event failures that result in core damage. The frequency of each sequence is determined by the event tree structure, the initiating event frequency and the Top Event split fraction frequencies specified by the RISKMAN master frequency file. RISKMAN allows the user to enter the split fraction names and the logic defining the split fractions (i.e., rules) to be selected for a given sequence based on the status of events occurring earlier in the sequence or on the type of initiating event.

### PSA Model Quality

The VYNPS PSA model is of sufficient quality and scope for this application. The VYNPS PSA model is highly detailed, including a wide variety of initiating events (e.g., transients, internal floods, LOCAs inside and outside containment, support system failure initiators), modeled systems, extensive level of detail, operator actions, and common cause events.

The current VY PSA model at the time of this analysis is VY04R1. The Level 1 and Level 2 VY PSA analyses were originally developed and submitted to the NRC in December 1993 as the

VYNPS Individual Plant Examination (IPE) Submittal. The VYNPS PSA model and documentation has been updated to reflect the current plant configuration, EPU design changes, and SBO/Vernon Tie evaluation as of September 2004. It also included the accumulation of additional plant operating history and component failure data.

The PSA model has been updated several times since the IPE due to the following.

- Equipment performance – As data collection progresses, estimated failure rates and system unavailability data change.
- Plant configuration changes – Plant configuration changes are incorporated into the PSA model.
- Modeling changes – The PSA model is refined to incorporate the latest state of knowledge and recommendations from industry peer reviews.

The VYNPS internal events received a formal industry PSA peer review in November 2000. All of the A and B priority comments have been addressed by Entergy and incorporated into the current VYNPS PSA model as appropriate.

PSA Update for COP Evaluation

The risk assessment examines the VYNPS PSA plant-specific accident sequences in which the containment integrity remains intact or the containment is impaired. Specifically, the following were considered:

- Core damage sequences in which the containment remains intact and NPSH is lost due to higher than expected torus temperatures.
- Core damage sequences in which containment integrity is impaired and NPSH is lost due to torus cooling failure and subsequent torus heatup.

The steps taken to perform this risk assessment are as follows:

1) Modify the VYNPS PSA Containment Isolation System fault tree to reflect the latest probability on the occurrence of pre-existing containment leakage.
2) Revise the appropriate LOCAs, FLOODS, ATWS, TRANSIENT event trees to reflect the impact of COP on NPSH requirements.
3) Perform an uncertainty analysis on a number of important basic events associated with the containment overpressure modeling changes used in this risk assessment.
4) Characterize the risk assessment evaluation of containment overpressure impact on NPSH requirements by change in CDF and LERF risk metrics.
5) Calculate the changes in CDF and LERF.

Details of the risk assessment, including quantitative results, will be subsequently submitted.

PSA Fault Tree and Event Tree Analyses

The specific issue to be assessed by this evaluation – control of containment over-pressure (COP) for ensuring adequate NPSH of RHR and Core Spray pumps – will be detailed in the PSA, to be subsequently submitted.

PSA Core Damage Impact

In order to assess the risk impact of utilizing containment accident pressure (containment overpressure) to satisfy the NPSH requirements for RHR and Core Spray pumps, the PSA model is being changed. The PSA model changes will be detailed in a subsequent submittal.

PSA Large Early Release Frequency Impact

In order to assess the risk impact of utilizing containment accident pressure (containment overpressure) to satisfy the NPSH requirements for RHR and Core Spray pumps, the PSA model is being changed. The PSA model changes will be detailed in a subsequent submittal.

Summary of PSA Evaluation Results

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10-6/ry and increases in LERF below 10-7/ry. It is anticipated that the proposed change (i.e., use of COP to satisfy the NPSH requirements for RHR and Core Spray pumps) will represent a very small incremental change in CDF and LERF, it is expected that the change will be non-risk significant from a risk perspective.

PSA Uncertainty

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

Uncertainty is categorized here into the following three types, consistent with PSA industry literature:

- Parametric
- Modeling
- Completeness

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

*Modeling uncertainty is focused on the structure and assumptions inherent in the risk model.* The structure of mathematical models used to represent scenarios and phenomena of interest is a source of uncertainty, due to the fact that models are a simplified representation of a real-world system. Model uncertainty is addressed here by the identification and quantification of focused sensitivity studies. The model uncertainty analysis and associated results are discussed further below.

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

**Parametric Uncertainty Analysis**

Sources and Treatment

Parametric uncertainty associated with the numerical results of this study primarily caused by insufficient component failure mode data, problems interpreting failure data and component performance records, the use of generic data in a plant-specific data analysis, and the intrinsic variability of failure data. In assessing the contribution of parametric uncertainty to the numerical results, the parameters of interest are those used by the accident-sequence logic models. They include initiating event frequencies, component failure rates and unavailability, and human error probabilities

In this study, parametric uncertainties were handled by defining a probability distribution for the value of each parameter such that the "nth" percentile of the distribution represents the value for which the analyst has n% confidence that the true value lies below the value. This subjective approach to the representation of uncertainty makes the propagation of parametric uncertainty through the evaluation mathematically straightforward. The evaluation was made using the Monte Carlo sampling technique. The uncertainty ranges characterized by the distributions vary in origin. For example, if the estimates are based on plant-specific data, the range is characteristic of statistical uncertainty. If the estimates are generic (or non-plant specific) the range is characteristic of the factors that may affect the failure properties of the component in different uses and environments. Hence the range will include plant-to-plant variation.

The propagation of uncertainties was accomplished using the RISKMAN computer program to calculate probability distributions and determine the uncertainty in the accident frequency estimate. The modeling of uncertainties and their propagation is discussed and documented in NUREG/CR-4550, Volume 1, Revision 1.

Quantification

The uncertainty of the parameter values is propagated through the PSA models. Quantitative results of the parametric uncertainty assessment will be provided in the subsequent submittal.

RISKMAN has three analysis modules: Data Analysis Module, System Analysis Module, and Event Tree Analysis Module. Appropriate probability distributions for each uncertain parameter in the analysis is determined and included in the Data Module. The System Module combines the individual failure rates, maintenance, and common cause parameters into the split fraction frequencies that will be used by the Event Module. A Monte Carlo routine is used with the complete distributions to calculate the split fraction frequencies. Event trees are quantified and linked together in the Event Module. The important sequences from the results of the Event Module are used in another Monte Carlo sampling step to propagate the split fraction uncertainties and obtain the uncertainties in the overall results.

The descriptive statistics calculated by RISKMAN for the total core damage frequency of the plant caused by internal events include:

- Mean of the sample
- Variance of the sample
- $5^{th}$, $50^{th}$, and $95^{th}$ percentiles of the sample.

**Modeling Uncertainty Analysis**

As stated previously, modeling uncertainty is concerned with the sensitivity of the results due to uncertainties in the structure and assumptions in the logic model. Modeling uncertainty has not been explicitly treated in many PRAs, and is still an evolving area of analysis. The PSA industry is currently investigating methods for performing modeling uncertainty analysis. EPRI has developed a guideline for modeling uncertainty that is still in draft form and undergoing pilot testing. The EPRI approach currently being tested takes the rational approach of identifying key sources of modeling uncertainty and then performing appropriate sensitivity calculations. This approach is taken here.

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

- Calculation of HEPs for other key actions
- External RPV injection credit
- Calculation of pre-existing containment degradation

Calculation of Pre-Existing Containment Degradation

An evaluation was performed to determine the maximum size hole in containment that would still assure adequate overpressure. Using the conservative 10CFR50 Appendix K containment analysis as the starting point, the maximum leak was determined to be approximately $27 \times L_a$. Allowable VYNPS integrated primary containment leakage $L_a$ is defined in the VYNPS TS Bases 4.7 as 0.8 wt % per day at 44 psig. The containment pre-existing leakage probabilities (basic events ISDWSMLEAK and ISDWLGLEAK) were determined using EPRI's TR-1009325.

Quantitative sensitivity cases to assess the response of CDF to changes in the allowable pre-existing leakage rate are being performed, where the probability value assigned to basic event ISDWSMLEAK is determined using EPRI's TR-1009325. The containment leakage rates evaluated in the sensitivity were $1 \times L_a$, $30 \times L_a$ and $60 \times L_a$. The maximum deterministic analysis leak rate of $27 \times L_a$ falls approximately in the middle of this range. The results of these sensitivity cases will be provided in the subsequent submittal.

**Completeness Uncertainty Analysis**

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

Seismic

The VYNPS seismic risk analysis was performed as part of the Individual Plant Examination of External Events (IPEEE). Entergy performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the seismic risk evaluation.

The conclusions of the VYNPS IPEEE seismic analysis are as follows:

For VYNPS, the SMA identified that the lowest HCLPF components in the selected primary and alternate safe shutdown paths are the Condensate Storage Tank (CST) with a HCLPF of 0.25g and the Diesel Fuel Oil Storage Tank (FOST) with a HCLPF of 0.29g. The HCLPF for all other components in the safe shutdown paths meet or exceed the 0.3g review level earthquake. These values represent significant margin to the design basis 0.14g earthquake.

The conclusions of the SMA are judged to be unaffected by the EPU or the containment overpressure credit issue. The EPU has little or no impact on the seismic qualifications of the systems, structures and components (SSCs). Specifically, the power uprate results in additional thermal energy stored in the RPV, but the additional blowdown loads on the RPV and containment given a coincident seismic event, are judged not to alter the results of the SMA.

The decrease in time available for operator actions, and the associated increases in calculated HEPs, is judged to have a non-significant impact on seismic-induced risk. Industry BWR seismic PSAs have typically shown (e.g., Peach Bottom NUREG-1150 study; Limerick Generating Station Severe Accident Risk Assessment; NUREG/CR-4448) that seismic risk is overwhelmingly dominated by seismic induced equipment and structural failures.

Based on the above discussion, it is judged that seismic issues do not significantly impact the decision making for the VYNPS EPU and containment overpressure credit.

Internal Fires

As discussed in the VYNPS EPU submittal, internal fires risk is not a significant contributor to the risk profile of the proposed EPU. Credit for containment overpressure is not required

## Element #5 – Monitored using Performance Measures

Containment Inerting

As a complimentary means for assuring the leak tightness and structural integrity of the VYNPS containment, during reactor power operation, the VYNPS containment is inerted with nitrogen (required by TS 3.7.A.7). Drywell pressure is maintained $\geq$ 1.7 psi positive (as required by VYNPS Technical Specifications 3.7.A.9) with respect to the suppression pool (torus) and consequently positive with respect to the outside atmosphere.

The time that the primary containment has less than 1.7 psi differential pressure is limited as follows, per Technical Specifications. Technical Specification 3.7.A.7 specifies that:

a. The primary containment atmosphere shall be reduced to less than 4 percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.7.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

c. If this condition cannot be met, an orderly shutdown shall be initiated immediately and the reactor shall be in a cold shutdown condition within 24 hours.

Technical Specification 3.7.A.9 on Drywell/Suppression Chamber d/P requires the following:

a. Differential pressure between the drywell and suppression chamber shall be maintained $\geq$1.7 psi except as specified in 3.7.A.9.b and 3.7.A.9.c below.

b. The $\geq$1.7 psi differential pressure shall be established within 24 hours of achieving operating pressure and temperature. The differential pressure may be reduced to <1.7 psi 24 hours prior to commencing a cold shutdown.

c. The differential pressure may be reduced to <1.7 psi for a maximum of four hours (period to begin when the $\Delta P$ is reduced to <1.7) during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-suppression chamber vacuum breakers, and the suppression chamber-reactor building vacuum breakers, and SGTS testing.

d. If the specifications of 3.7.A.9.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

The differential pressure is alarmed and recorded per Technical Specifications. Technical Specification 4.7.A.9 requires:

a. The differential pressure between the drywell and suppression chamber shall be recorded once per shift.

b. The operability of the low differential pressure alarm shall be verified once per week.

for VYNPS Appendix R fire accident sequences. As such, it is judged that fire issues do not significantly impact the decision making for the VYNPS EPU and associated containment overpressure credit

Other External Hazards

In addition to seismic events and internal fires, the VYNPS IPEEE Submittal analyzed a variety of other external hazards:

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

The VYNPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that VYNPS meets the applicable NRC Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards. As such, these other external hazards are judged not to significantly impact the decision making for the VYNPS EPU and containment overpressure credit.

Note that the VYNPS IPEEE also analyzed internal flooding scenarios. However, internal flooding scenarios are now incorporated into the current VYNPS PSA internal events model of record.

Shutdown Risk

As discussed in the VYNPS EPU submittal, shutdown risk is a non-significant contributor to the risk profile of the proposed EPU. The credit for containment overpressure is not required for accident sequences occurring during shutdown. As such, shutdown risk does not influence the decision making for the VYNPS EPU containment overpressure credit.

Verification of the required positive pressure differential is made and recorded during each shift (as required by VYNPS Technical Specifications 4.7.A.9.a). In addition, a control room alarm will annunciate upon low differential pressure. The torus air space pressure is typically slightly positive with respect to the atmosphere (about 0.1 psig). Although normal operating pressures in the drywell and torus atmosphere are less than that resulting from a design basis accident, the fact that the containment is pressurized provides a reliable means of verifying that no large leak paths exist in the containment structure. Specifically, any substantial containment leak path will cause operational difficulties in maintaining positive pressure in the containment and the condition will be self-evident by manifesting itself in an excessive nitrogen make-up rate. Monitoring for containment leakage is accomplished by monitoring the average daily nitrogen consumption used by the containment inerting system and is determined daily by the performance of plant surveillances. Significant containment leakage would be identified through increased nitrogen usage needed to maintain the required TS pressure, and would be investigated promptly and addressed within the scope of the plant corrective action program.

<u>MSIV Leakage</u>

Technical Specification 3.7.A.4 states:

1. Whenever primary containment integrity is required:
   a. The leakage rate from any one main steam isolation valve (MSIV) shall not exceed 62 scfh at 44 psig (Pa)
   b. The combined leakage rate from the main steam pathways shall not exceed 124 scfh at 44 psig (Pa)

La is equivalent to 320.2 scfh. Therefore, the combined allowed leakage rate of 124 scfh is equivalent to only 0.39 La which is a small percentage of the leakage rate required to lose COP.

Therefore, the TS allowed combined leakage from the main steam pathways will not result in a less of COP.

<u>Component Testing</u>

Per Technical Specification 4.6.E:

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a (g), except where specific written relief has been granted by the NRC.

   Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter or in accordance with alternate measures approved by NRC Staff.

2. Operability testing of safety-related pumps and valves shall be performed in accordance with the Code of Record as required by 10 CFR 50.55a, except where specific written relief has been granted by the NRC.

Primary Containment Isolation Valves are tested per T.S. 4.7.D

1. Surveillance of the primary containment isolation valves should be performed as follows:
   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure time at least once per operating cycle.
   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

Containment Leakage Testing

VYNPS currently applies a performance-based testing program. Technical Specification leakage limit criteria specify actions if exceeded. In accordance with TS 6.7.C, the integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests, and the overall leak-tight integrity of the primary containment is verified periodically by a Type A test (integrated leakage rate test) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak-tight characteristics of the primary containment at the design basis accident pressure. The change in licensing basis to support VYNPS operation at EPU conditions does not alter the basic methods by which Appendix J leakage rate testing is performed, or the frequency of testing, or acceptance criteria.

A number of surveillances are periodically conducted to ensure the integrity of the primary containment function, including the maintenance of an inert atmosphere under positive pressure during reactor power operation. Testing frequencies are specified in the plant Technical Specifications and are performance-based in accordance with 10 CFR 50, Appendix J, Option B.

Maximum Containment Opening Size

An evaluation was performed to determine the maximum size hole in containment that would continue to ensure adequate overpressure, and determine if that size would be detectable during normal operation.

The maximum size hole in the containment that could be tolerated and still assure adequate overpressure for NPSH margin was determined. Using the conservative design basis LOCA containment analysis as the starting point, the maximum leak was determined to be equivalent to a 0.475 inch diameter opening in the drywell assuming a loss coefficient (K) of 1.5. This was determined using a GOTHIC model that was benchmarked against the GE containment analysis results.

$L_a$ is defined in the VYNPS TS Section 4.7 Bases as 0.8 wt % per day at 44 psig. This TS allowable value is converted to a maximum allowable leakage rate of 24.076 lbm/hr, or 320.210 scfh (assuming standard T and P defined as 68 degrees F and 14.696 psia, respectively). Using Cranes standard, and assuming K = 1.5, an equivalent hole diameter that corresponds to $L_a$ was determined to be 0.091 inches. Therefore, the leakage rate from the maximum diameter opening determined above would be equal to $(0.475/0.091)^2$ times $L_a$, since flow rate is proportional to the square of the diameter, all other things being equal. Thus, the maximum leakage rate that could be tolerated under design basis conditions is $27xL_a$, or approximately 8,645 scfh at 44 psig and 68°F.

This size hole would be detectable during normal operation. The drywell operates with a positive pressure differential between the drywell and torus. The differential is approximately 1.7 psi, which is equivalent to the amount of pressure required to depress the water column in the downcomers. The torus is vented to the SBGTS via a normally open 3-inch torus vent bypass valve (AC-6B). Thus, the torus is always maintained at essentially atmospheric pressure. Except for situations where the pressure differential is disturbed by normal plant maneuvers, the differential is maintained by normal in-leakage to the drywell from instruments and other equipment supplied by nitrogen from the nitrogen supply system. This in-leakage is monitored by VYNPS operators on a twice-daily basis and is typically on the order of 300 scfh.

If drywell leakage to the reactor building were to increase during normal operation, the ability to maintain the normal drywell-to-torus pressure differential, which is a TS requirement, would be compromised and operators would be required to investigate the cause. For example, a leak equivalent to or greater than a 0.21 inch diameter hole (K=1.5) would require more than 300 scfh to maintain a differential pressure of 1.7 psid and would be noticed as an anomaly by the operators. The leakage from a 0.21 inch diameter hole represents less than 20% of the maximum leakage that could be tolerated and still ensure adequate overpressure for ECCS pump NPSH margin.

<u>Containment Leakage Summary</u>

1. Containment leakage 27 times greater than current allowable ($L_a$) could be tolerated without compromising overpressure required to ensure adequate NPSH margin.

2. Drywell leakage that is a small fraction of the maximum that could be tolerated for NPSH margin would be easily detected as an anomaly on drywell-torus pressure differential requirements during normal operation and action could be taken to address the issue.

<u>Practicality of Plant Modifications</u>

Entergy has not fully evaluated what physical, operational and analytical changes to the VYNPS would be necessary to eliminate the need to take credit for COP, but the changes would in any case be quite substantial and would include significant changes to plant systems, structures and components. For example, some of the modifications under consideration include replacement of the ECCS pumps and/or replacement of the RHR heat exchangers. Discussions with the VYNPS ECCS pump manufacturers resulted in the conclusion that the existing pumps and motors could not be modified in order to eliminate the need for COP. A new pump design would be required. Alternately, the RHR heat exchangers would likely be of a new design in order to accommodate the significant increase in capacity required to eliminate the need for COP.

Replacement of six ECCS pumps or two RHR heat exchangers could result in the following potential impacts to systems, components, and structures:

- Replacement of RHR and CS pump/motor support pads and anchors;
- Piping modifications to allow connection to the new RHR and CS pump nozzles or new RHR heat exchanger nozzles;
- Revised structural equipment supports;

Pump and motor or heat exchanger replacements would require the preparation of engineering modification packages to facilitate the hardware changes. The modification packages would include extensive rigging and scaffolding necessary to facilitate the removal of existing equipment and the installation of new equipment. Because one RHR heat exchanger, two RHR pumps, and one CS pump are located in each of two tightly spaced cubicles, replacement of the pumps/motors would be performed sequentially, not in parallel. Furthermore, since RHR cooling capability must be maintained during shutdown, replacement of the RHR pumps/motors or heat exchangers in the two loops would also be performed sequentially, which would effectively double the length of the outage. One alternative would be to install the replacement equipment over two or more outages.

The pump/motor or heat exchanger replacements could result in the following potential secondary impacts:

- Revision of stress and supports analyses due to equipment nozzle piping modifications;
- Revision of Electrical System Studies including EDG load analysis based on new, slower operating pump motors;
- Reanalysis of motor protection relay settings;
- Revision of LOCA analysis based on new, slower operating pump motors and results of revised EDG load analysis;
- Revision of various plant design analyses (for example: Service Water hydraulic analysis or Motor Operated Valve analyses), that might be affected by changes to pump or heat exchanger parameters.
- Temporary or permanent relocation of interfering commodities to facilitate the removal of existing equipment and the installation of new equipment.