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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 163 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

The Vermont Yankee Nuclear Power Station (VY) is a boiling water reactor (BWR), model BWR-4, with a Mark I containment. By letter dated May 8, 1998, as supplemented on July 10, 1998 and October 2, 1998, the Vermont Yankee Nuclear Power Corporation, the licensee for the Vermont Yankee Nuclear Power Station, submitted for Nuclear Regulatory Commission (NRC) staff review a proposed change to the technical specifications (TS). The proposed TS amendment would change the normal operating suppression pool water temperature limit from 100 °F to 90 °F with an allowance for the suppression pool temperature to rise to 100 °F for up to 24 hours during a surveillance which adds heat to the suppression pool. The original TSs approved for the plant specified a 90 °F limit; however, amendment #88 changed the TSs to operate with the suppression pool temperature up to 100 °F during all normal operational modes. A later review performed by the licensee identified that all elements of the change had not been properly incorporated and reanalysis of the containment and reactor response was performed. Reducing the peak suppression pool temperature back to 90 °F during normal operation will provide consistency between the final safety analysis report (FSAR) conditions used in the Chapter 15 analyses for licensing the plant and the TSs. A suppression pool temperature of 90 °F was used as an input for the containment analyses. The October 2, 1998, supplement did not affect the initial no significant hazards determination.

The changes proposed are as follows:

1) T.S. 3.7.A.1.a, maximum water temperature during normal operation: Change from 100 °F to 90 °F.

2) T.S. 3.7.A.1.b, maximum water temperature during any test operation which adds heat to the suppression pool: The 100 °F limit will be retained with the addition of a restriction that the temperature shall not be above the normal operating limit for more than 24 hours.

3) TS 3.7.A.1.c, change the suppression pool temperature at which power operation is allowed to resume from 100 °F to 90 °F.

9901080007 981228
PDR ADOCK 05000271
P PDR

2.0 EVALUATION

As a basis for the temperature limit on torus water temperature the licensee stated that a suppression pool temperature of 90 °F was used as an input for the containment analyses with acceptable results. An initial suppression pool temperature of 90 °F results in a peak post accident torus water temperature of less than 185 °F. The containment analyses were done for loss-of-coolant-accident (LOCAs) and events involving safety relief valve (SRV) discharges to the suppression pool. The licensee stated that environmentally qualified electrical equipment was determined to be qualified for the expected temperatures, including 185 °F for the torus temperature. In addition, instrument accuracy and ECCS pipe stress were also evaluated for the effect of a 185 °F torus temperature with no adverse effects. The licensee stated that all safety analysis requirements are met with a normal operating limit of 90 °F.

The licensee's analysis described the methodology used to calculate the maximum suppression pool temperature during the analyzed accident scenarios which reject heat to the suppression pool. The TS suppression pool temperature limits were derived from RELAP5YA-B1A and GOTHIC 5.0e accident analysis of the suppression pool heatup following various heat rejection load accident scenarios. These scenarios used conservative assumptions and methods for a design basis accident LOCA (DBA-LOCA) with limiting single failures to maximize the heat rejection load to the suppression pool following the accident. Additionally, individual sensitivity runs to increase the short- and long-term peak pool temperature were performed. These sensitivity studies utilized case-specific conservative assumptions to investigate the effect of varying different parameters during the limiting single failure cases. The new DBA-LOCA analysis performed to determine pool heatup incorporated the ANS 5.1 1979 decay heat model, increased residual heat removal (RHR) heat exchanger fouling in the RHR model, and included additional heat addition from the feedwater system. The ANS 5.1 1979 decay heat model has been found to be acceptably conservative. The two other models added additional conservatism to the analysis in comparison to the assumptions used in the original design basis by increasing the heat addition to the pool and decreasing the heat removal rate by the RHR. The most limiting short- and long-term peak pool temperatures were obtained from the RHR heat exchanger failure in combination with maximum ECCS injection flow or increased feed flow rate sensitivity studies, respectively.

NRC review of the proposed change focused on the containment response methodology, codes used, benchmarking of the computer codes, resulting containment response, effect on fuel clad integrity, and adequacy of net positive suction head (NPSH) for appropriate pumps.

2.1. Methodology, Codes, Benchmarking, and Resulting Containment Response

The licensee performed a detailed validation and benchmarking of the two computer codes to demonstrate the applicability of the approach. The containment response methodology consists of two distinct elements identifiable by the two computer codes used. They are:

a. LOCA mass and energy calculations using RELAP5YA-B1A code. A plant model was derived from the current NRC-approved LOCA licensing analysis per 10 CFR 50 Appendix K. The NRC Standard Review Plan (SRP) accepts the mass and energy release analysis for postulated LOCA events for use in containment analysis if the analysis complies with the relevant requirements of Appendix K paragraph I.A. In general, these requirements assure that the approved Appendix K evaluation model is used with appropriately conservative inputs for containment analysis. This calculation modifies the NRC-approved VY LOCA Appendix K model with inputs chosen to conservatively calculate suppression pool temperature and wet-well pressure and meets many of the requirements detailed in the SRP. Changes from the Appendix K analyses include maximizing the vessel inventory by bounding the initial water level in the downcomer to the maximum expected normal operating level, setting the power level at 1625 Mwt, basing the mass and energy release on 107 percent flow, and performing a sensitivity study to assure that all ECCS flows are conservatively established.

The staff has reviewed this methodology and agrees with the licensee that performing the mass and energy releases in accordance with Appendix K paragraph I.A is appropriate for containment analyses. In addition, the changes to the Appendix K analysis, as noted above, are also appropriate. In combination, the staff has concluded that the methodology as proposed for establishing the mass and energy release profiles to support the proposed TS changes is acceptable with feedwater addition considered in a conservative manner as discussed below.

b. Containment calculation using the GOTHIC 5.0e code. GOTHIC is used in this calculation to perform the dynamic mass and energy balance on the containment. It has been validated against a selected matrix of separate effects and integral tests to evaluate the available modeling choices. More importantly, benchmarking is included in this calculation for the purpose of demonstrating a direct comparison of results to similar results previously found to be acceptable by the NRC.

The NRC concluded that the basic methodology was reasonable based on conformance with Appendix K paragraph I.A. and the benchmarking described later in this evaluation.

The overall shutdown process was also reviewed. It was found that the consideration of feedwater addition was essential in establishing a conservative analysis. A review found that after a large-break LOCA or main steamline break, the operators were likely to use continued feedwater in order to assist in mitigation and recovery of the accident. As a result, the developed methodology used in the calculations assumes conservative feedwater injection from the perspective of maximizing suppression pool temperature. Continued feedwater addition until the incoming water temperature is less than the expected peak suppression pool temperature is considered conservative. For this purpose a 175 °F value was picked to terminate feedwater flow. The selection of 175 °F was based on the assumption that this represents the approximate peak suppression pool temperature. However, the most recent reanalyses were showing results higher than the 175 °F selected value. Therefore, a sensitivity study was performed. It showed that an increase above 175 °F, but below 185 °F, would have a negligible impact on the resulting peak suppression pool maximum temperature. Based on this result, the licensee concluded that an iteration was unnecessary and that the 175 °F value remained an adequate feedwater shutoff temperature for analytical purposes. The staff agrees with this assessment since it results in a negligible impact on the resulting peak suppression pool maximum temperature.

The initial mass and energy release profile was mechanistically calculated using a detailed RELAP5YA model of the reactor vessel with a coupled feedwater system model for the initial blowdown. As discussed below, this initial time period was established as the first 80 seconds. After 80 seconds, a single node vessel model with a coupled feedwater model is used with appropriate decay heat and passive metal structure heat transfer and incorporated into the GOTHIC code. The containment response to the mass and energy release is based on simple mass and energy balances on the drywell and wetwell.

The coupled feedwater model assumes a constant feed flowrate for the analysis. The adequacy of this assumption is dealt with by a sensitivity evaluation for a range of feedwater flowrates. At 80 seconds, it is calculated that the reactor pressure vessel (RPV) and ECCS conditions stabilize to quasi-steady state and drywell pressures equalize and the core power output is essentially decay heat. The GOTHIC code has the capability to model heat removal from the suppression pool using a dynamic heat exchanger model as well as modeling the RHR and Core Spray system interaction with the reactor vessel.

The selection of the 80-second transition time was based on the results that a quasi-steady state set of conditions existed in both the reactor vessel as well as the containment. In addition, the RELAP results showed flow reversals at times greater than 100 seconds. While the occurrence of this calculated short-term reverse flow is physically plausible, the calculation of the effects of reverse flow in the RELAP model is not reliable due to the lack of mechanistic modeling of the effects of non-condensables. In order to avoid the use of RELAP results after non-condensables would be introduced into the vessel, the transition to the GOTHIC model is made prior to the prediction of reverse flow in the RELAP model.

An important aspect of obtaining a conservative analysis is to employ a reasonably conservative decay heat model. For both codes, the decay heat is calculated using ANS 5.1-1979 decay heat standard considering the 2% calorimetric uncertainty and a 2 sigma uncertainty. The staff has concluded that the ANS standard using the two uncertainties identified above is an acceptable decay heat model since it is based on the best available data and the two major uncertainties have been adequately considered.

Benchmarking was used to address why these two computer codes are appropriate for use in performing licensing based containment analyses. These benchmarks have been performed specifically to assess the adequacy of the methods and models used in the calculations. With respect to the RELAP code, separate effects benchmarks for the RHR heat exchanger and feedwater model were performed. These assessments were made against plant surveillance criteria and plant trip data. The benchmarks provide the basis for judging the adequacy of both of these models. Additionally, the comparisons demonstrate that the models are conservative and account for uncertainties and therefore are adequate. RELAP was also benchmarked against the DBA analysis documented in the FSAR. Good agreement was established during the first 70 seconds of the blowdown. Much beyond this time period, flow oscillations were seen to occur in the RELAP output. It was concluded that the oscillations were not representative of the expected conditions, therefore, only the results prior to the development of the oscillatory flow patterns should be considered for comparative purposes. Note that the RELAP results for the reanalysis did not show flow oscillations well beyond 100 seconds. However, when modeled against the DBA analysis a much earlier time of 70 seconds was observed. The time difference between the two analyses demonstrates that modeling has an impact on when the flow instabilities will be computed. Therefore, the staff considers it acceptable to use these results since the two computer codes used demonstrate excellent agreement with the DBA analysis documented in the FSAR for times prior to any flow oscillations.

For the containment analyses using GOTHIC, a comparison of the integrated method against the DBA analysis documented in the FSAR was performed. This comparison provides a total method and model comparison against a previously NRC-approved analysis. The results showed that the proposed approach is adequate for torus temperature and pressure analysis, and the overall modeling technique yields conservative results.

Having established reasonable agreement with the comparisons against the FSAR analysis, a sensitivity study was performed to ensure that the various key parameters are sufficiently conservative. To initiate this study, a base case LOCA analysis was performed. This case was selected as being reasonable and representative of the spectrum of possible sequences. Then a series of eight cases were considered with each case representing a variation of a single parameter. The results showed that while the base case was not the most bounding case, it was representative, and therefore acceptable for use in the analysis. The most significant variation in the peak suppression pool temperature was 8 °F higher than the base case. As a result, maximum suppression pool temperature will remain below 185 °F.

Through benchmarking and sensitivity studies, VY has been able to show that the use of the RELAP5YA-B1A and GOTHIC 5.0e codes for this specific application will produce conservative results. The staff finds, for the reasons set forth above, the methodology and code applications to be acceptable.

2.2 Effect on Fuel Clad Integrity

The staff reviewed the information contained in the submittal describing the relationship between the temperature of the suppression pool and fuel clad integrity. The licensee stated that the analyses performed in accordance with 10 CFR 50.46 demonstrate the core cooling capability of the ECCS. The analyses assume that the RHR and CS pumps take suction from the suppression pool. The analyses are based on approved methods that are independent of the initial and transient suppression pool temperatures. Since the suppression pool temperature does not effect the fuel clad integrity, the proposed change and the supporting technical evaluation have no adverse impact on the 10 CFR 50.46 analysis results.

The staff reviewed the determination that the calculated suppression pool temperature is independent of the ability to cool the core. This independence occurs because the methodology used to calculate peak clad temperature assumes that all of the core spray cooling flow passes down through the bypass region without removing heat from the core region and drains into the lower plenum. In the lower plenum, the water heats up and boils from the heat contained in the plenum components. The steam produced flows upward through the core to provide fuel cooling. This is the only mechanism that removes heat from the fuel until the counter current flow limit is no longer in effect. These assumptions are introduced through the ECCS models and are considered to be conservative since this increases the heat in the core and results in a more limiting peak clad temperature. The staff-approved methodology for obtaining peak clad temperature using the General Electric SAFER/GESTER code was approved by the NRC for use by VY for its LOCA analysis. Using this methodology, the more conservative temperature for the cooling water is a lower value because colder water takes longer to heat up and boil before providing steam cooling for the core. The original SAFER/GESTER analysis for VY fuel clad integrity assumed an emergency water supply temperature of 120 °F. The RELAP/GOTHIC analysis determined that the containment suppression pool which supplies water to the emergency core cooling system will eventually rise to slightly less than 185 °F. Therefore, the results of this calculation are bounded by the original fuel clad integrity analysis and are acceptable to the staff. The staff also notes that a telephone discussion with the licensee was conducted to clarify the submittal information.

2.3 NPSH Considerations

The NPSH calculations were performed for the core spray (CS) and RHR pumps using the short- and long-term peak pool temperatures. These calculations demonstrated that the most limiting condition was the CS pump long-term peak temperature NPSH calculation which has a safety margin of 0.5 feet of water between the available and required values. The most limiting condition for the RHR pump is also the long-term peak temperature which resulted in a 1.7 foot margin. In both calculations, available NPSH was greater than required NPSH; thus, the pumps would function as required and provide core cooling during the accident scenarios analyzed. The methodology used for determining NPSH and the values obtained from the calculation were submitted on July 31, 1998, in the VY response to GL 97-04 and approved for use by the staff.

The staff reviewed the assumptions used in the RELAP analysis to obtain the maximum temperature of the suppression pool, the calculations for the available NPSH, and the evaluation of available versus required NPSH for pump operability. The staff determined that the analysis performed to demonstrate that available NPSH is greater than required NPSH and therefore is acceptable for the short- and long-term peak suppression pool temperatures attained with the maximum heat load rejection to the pool. The calculations demonstrated that the most limiting NPSH requirement of the CS pump during the long-term peak temperature was met with an additional 0.5 foot margin provided in available NPSH. Additionally, the CS pump seals have an operational range of 32 °F to 210 °F so they will not be affected by the temperature of the pool and the RHR pump seal is capable of operating with the fluid temperature at 185 °F for one week with negligible effect on seal life. The licensee indicated that the suppression pool temperature is calculated to be above 180 °F (and below 185 °F) for less than 12 hours, which is well within the capability of 185 °F for one week. Therefore, the staff has concluded that the analysis for lowering the normal operating suppression pool temperature is acceptable since approved methodologies for the fuel clad integrity and NPSH were used, and applicable limits are maintained.

2.4 Summary

The proposed change to TS 3.7.A.1.a and TS 3.7.A.1.c to reflect a maximum water temperature of 90 °F for the suppression pool is acceptable since all safety analyses requirements are met using this input temperature. The proposed change to TS 3.7.A.1.b to specify a 100 °F limit on torus water temperature following testing for a time not to exceed 24 hours allows for appropriate testing of safety-related equipment to ensure operability and is therefore acceptable.

In summary, based on our review the staff concludes that the proposed Vermont Yankee Nuclear Power Station TS changes are acceptable. The changes return the temperature limits to the values that were in place prior to the modification to the TSs by amendment 88.

The supporting analyses now fully support the suppression pool temperature values. The latest analyses employ different computer codes for both mass and energy release and containment response. These codes are RELAP5YA-B1A and GOTHIC 5.0e. Through benchmarking and sensitivity studies, VY has been able to show that the use of these codes for this specific application will produce conservative results. Therefore, the staff finds the methodology and code applications to be acceptable.

In addition, the staff concludes that the proposed Vermont Yankee Nuclear Power Station Technical Specification change is acceptable because it reduces the allowed suppression pool temperature to a value which provides for adequate long term NPSH following the accident which provides the greatest suppression pool heatup. This analysis was performed for the CS and RHR pumps using the maximum peak pool temperatures. The proposed change is also acceptable since fuel integrity was shown to be maintained using the NRC-approved code for LOCA analysis, SAFER/GESTER.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 50941). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Kudrick and U. Shoop

Date: December 28, 1998