DE&S
Duke Engineering & Services

Vermont Yankee Nuclear Power Station
Peak Suppression Pool Temperature Analyses
for Large Break LOCA Scenarios
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1.0 INTRODUCTION

This report provides the results from an evaluation of Vermont Yankee (VY) suppression pool temperature response to a DBA-LOCA that has been performed by DE&S for Vermont Yankee Nuclear Power Corporation (VYNPC). The evaluation included sensitivities to determine the limiting short-term and long-term suppression pool temperature for use in determining the limiting available net positive suction head (NPSH) for the residual heat removal (RHR) and core spray (CS) pumps. The short-term response occurs over the first ten minutes after the start of the LOCA event during which no credit is taken for operator actions to throttle pump flow or initiate containment cooling. The long-term response is the period after ten minutes when it is assumed that the operators control pump flow and initiate containment cooling.

The current VY licensing basis for suppression pool temperature is a double-ended guillotine (DEG) suction line break (i.e. DBA-LOCA) detailed in the Vermont Yankee FSAR Section 14.6.3. In addition to the DBA-LOCA, suppression pool temperature response to other events such as small break LOCA, main steam line break (MSLB), Appendix R transients, anticipated operational transients (AOT), and station blackout are required to be analyzed to ensure all containment design criteria are met. The analysis initial conditions and results provide the basis for the Technical Specification on initial torus temperature and the limits on suppression pool water volume.

The methodology developed and used in the evaluation consists of a blowdown mass and energy release calculation based on an approved 10CFR50 Appendix K LOCA model using the RELAP5YA-B1A computer code and a post-blowdown mass and energy release and suppression pool temperature response calculation using the GOTHIC 5.0e computer code. Conservative input assumptions have been used to ensure the calculated peak suppression pool temperature is maximized.

The effects of several emergency core cooling system (ECCS)[1] single failures in conjunction with a large break LOCA were examined. The single failures that were analyzed included: 1) loss of the DC-1 bus, 2) loss of an RHR heat exchanger, and 3) loss of a DC-1 bus but no pump failures due to room heatup. For these events, normal power was assumed to be available which allowed

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[1] The Vermont Yankee FSAR uses the term "core standby cooling system" (CSCS). In this report, the two terms, ECCS and CSCS, are interchangable.

hot feedwater addition. Several sensitivities were evaluated including: 1) recirculation line discharge valve operation, 2) RHR and CS pump injection flow rates, and 3) feed injection flow rate. The results presented in this report can be used by VYNPC to calculate the available NPSH margin for pumps taking a suction from the suppression pool as well as to evaluate other plant design features that might be affected by peak suppression pool temperature.

A benchmark analysis was performed to validate the methodology used in the evaluation. The results of the benchmark analysis were compared to the results provided in the Vermont Yankee FSAR (Section 14.6.3) for the DBA-LOCA, the current licensing basis for Vermont Yankee. The comparison demonstrates the impact on the suppression pool temperature response of utilizing a more detailed analysis methodology.

2.0 METHOD DESCRIPTION

The method developed for the evaluation models the major plant systems that have an impact on post-LOCA suppression pool temperature response. Only safety-related systems are credited for mitigating effects. Non-safety systems, such as condensate and feedwater, are considered only when the effects could increase suppression pool temperature or decrease available RHR and CS pump NPSH.

The major systems modeled in this analysis are:

- reactor vessel and coolant system
- reactor feedwater system
- primary containment
- low pressure coolant injection / residual heat removal system (LPCI / RHR)
- core spray system (CS)
- RHR service water system (RHRSW)

Figure 1 provides a schematic of the interface of these different systems. During a large break LOCA, the reactor vessel will depressurize and reactor coolant will be discharged to the primary containment. The reactor feedwater system may continue to deliver hot feedwater to the reactor vessel at the discretion of the operators dependent on availability and procedural guidance. Upon the proper actuation signal, the LPCI and core spray system pumps will start. After the reactor vessel pressure drops below the pressure permissive, the injection valves for these two systems will open allowing water from the suppression pool to be delivered to the reactor vessel.

After the operators are able to diagnose the situation the RHR service water pumps can be manually started allowing heat removal through the RHR heat exchangers. This aids in core cooling and removes heat from the primary containment delivering it to the ultimate heat sink. The RHR system may be realigned allowing direct water return to the suppression pool after passing through the RHR heat exchanger. Alternatively, the operators may continue with the RHR system injecting into the vessel in LPCI mode through the RHR heat exchanger.

The operators may be directed by procedure to initiate drywell or wetwell spray. Drywell and wetwell spray is initiated manually by realigning valves which direct RHR flow out of the RHR heat exchanger to spray headers in the drywell and wetwell.

The containment analysis methodology consists of two distinct elements identifiable by the computer code each element requires. The two elements are:

a. LOCA blowdown mass and energy release calculation using the RELAP5YA-B1A code and a plant model derived from the current NRC-approved LOCA licensing analysis per 10CFR50 Appendix K. The NRC-approved VY LOCA Appendix K model inputs were modified to conservatively calculate suppression pool temperature.

b. Containment calculation using the GOTHIC 5.0e code. GOTHIC was developed under EPRI and nuclear industry guidance specifically for containment analyses. 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. Additional benchmarking was performed for the purpose of validating specific models used in the evaluation.

To implement this methodology, the evaluation used specific steps leading to completion:

1. identify potentially limiting large break LOCA scenarios based on a detailed examination of potential single failures, availability of offsite power, and operator actions,

2. develop a base analysis model for analyzing the identified scenarios,

3. benchmark the analysis models to assure adequacy of both the models and the method,

4. analyze a base case scenario using the tested models and method,

5. analyze other scenarios and sensitivities to assist in understanding of the phenomena and scenario variabilities.

The scenario development process includes examining previous design basis calculations for post-LOCA torus temperature and ECCS single failure criteria analyses as well as the evaluation of balance of plant (BOP) systems, specifically feedwater and condensate. After a large break LOCA or MSLB, the operators, based on procedural guidance and equipment availability, may use feedwater in order to assist in mitigation and recovery from the accident. Alternatively, the feedwater system may continue to inject relatively hot water into the vessel if offsite power is not lost and no action is taken to prevent its addition. The feedwater system has the potential to add significant amounts of mass and energy to the primary containment. While beneficial from the perspective of ensuring continued cooling of the core, this can contribute significantly to the post-accident heatup of the suppression pool. The methodology developed and used in the evaluation assumes conservative feedwater injection from the perspective of maximizing suppression pool temperature.

The mass and energy release from the reactor vessel to the primary containment is mechanistically calculated using a detailed RELAP5YA model of the reactor vessel with a coupled feedwater system model. This detailed model calculates the mass and energy release until the reactor pressure vessel (RPV) and ECCS conditions stabilize to quasi-steady state, the RPV and drywell pressures equalize, and the core power output is essentially decay heat (about 80 seconds). This mass and energy release is input into a GOTHIC model of the Vermont Yankee primary containment. This containment model calculates the suppression pool temperature response to the mass and energy release to the quasi-steady state time.

After the quasi-steady state condition is reached, a less detailed vessel model is required due to the quasi-steady state nature of the RPV. The mass and energy release is calculated using a simple vessel model with appropriate decay heat and passive metal structure heat transfer with the GOTHIC code. The containment response to the mass and energy release is based on simple mass and energy balances on both the drywell and wetwell. The GOTHIC code is used to calculate this mass and energy balance. GOTHIC has the capability to model heat removal from the suppression pool using a dynamic heat exchanger model as well as modeling the RHR and CS system interaction with the reactor vessel. The GOTHIC model also includes structural heat sinks in the wetwell but not the drywell. Negligible heat is transferred from the containment to the environs other than via the RHR heat exchangers in the GOTHIC model.

A DBA-LOCA with various failures of mitigating equipment and with and without off-site AC power available are considered to determine the limiting event and conditions which are analyzed. Input parameters are at their conservative limiting value within the operating range, including uncertainties where applicable. Several types of single failures are evaluated - failures that produce degraded containment heat removal (e.g. loss of an RHR heat exchanger) and failures that degrade ECC injection capability affecting the rate at which heat is transferred from the vessel to the suppression pool (e.g. loss of a core spray pump). The rate of mass and energy addition to the vessel from the feed and condensate system is also considered.

The break analyzed is a DEG break at the location of the recirculation pump suction with off-site power available. This break location results in significant mass and energy being transferred to the containment from the vessel. The availability of off-site power allows the operators to continue hot feedwater injection into the vessel in an effort to maintain the core cooled. The hot feedwater contributes to the temperature increase of the suppression pool because the temperature of the incoming feedwater is higher than the temperature of the suppression pool. A variety of single failures were analyzed as well as ECCS operating modes to assess the limiting scenario conditions.

Uncertainties for the limiting breaks and single failures were addressed. These include uncertainties in the predictive tools (e.g. prediction of break flow rate, heat transfer rates in a heat exchanger, interfacial heat and mass transfer, etc.) and uncertainties in input parameters due to measurement inaccuracies, instrument drift, manufacturing tolerances, etc.

2.1 Code Description

RELAP5YA provides a consistent, integral analysis capability of the system and core response to LOCA events and other plant transients. Extensive assessments of RELAP5YA, against many separate effect and integral test results, have been performed. The Vermont Yankee vessel model is based on an approved 10CFR50 Appendix K ECCS evaluation model. The RELAP5YA assessments and the licensed model establish the viability of the RELAP5YA code and model to predict complex thermal-hydraulic phenomena such as those encountered in the reactor system analysis of LOCA events (Reference 1).

GOTHIC is a general purpose thermal-hydraulic code for design, licensing and operating analysis of nuclear power plant containments and other confinement buildings (Reference 2). GOTHIC solves the conservation equations for mass, momentum and energy for multi-component, multi-phase flow. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer. The interface models allow for the possibility of thermal non-equilibrium between phases and unequal phase velocities.

A simplified set of conservation equations are used in this calculation. All volumes in the GOTHIC models are lumped parameter (as opposed to subdivided, multi-dimensional) volumes. As such, the mass and energy balances are maintained among the volumes, and the interconnecting junctions pass the flow from one volume to the next. As the fluid flow in the GOTHIC models are primarily (post-blowdown) gravity-dominated flows, and the piping systems for the connecting systems are not modeled explicitly; the momentum balance across the junctions is of secondary importance. The GOTHIC models serve to solve the simple energy balance among the drywell, wetwell, vent system, and reactor vessel control volumes and between the liquid and vapor phases in each control volume. The GOTHIC assessments and model benchmarks establish the appropriateness of the GOTHIC code and models to predict the suppression pool temperature response to the LOCA events contained in the evaluation.

2.2 Benchmarking

A set of benchmarks have been performed specifically to assess the adequacy of the methods and models. Separate effects benchmarks for the RHR heat exchanger and RELAP feedwater model have been performed. The RHR heat exchanger model has been assessed against plant surveillance criteria for a variety of conditions to assure conservative modeling over the anticipated range of operating conditions. The feedwater model has been assessed against April 1997 plant trip data. These benchmarks provide the basis for judging the adequacy of the RHR heat exchanger model and the feedwater model. The benchmarks show that both of these models are conservative and account for relevant uncertainties and are, therefore, adequate.

A comparison of the integrated method against the DBA-LOCA analysis in the VY FSAR has been performed. The case corresponds to a DBA-LOCA without offsite power available and one operable RHR heat exchanger, one operable RHR pump, and one operable RHPSW pump. The benchmark provides a total method and model comparison against a previously NRC-approved analysis.

The separate effects benchmarks as well as the mass and energy balances performed during the evaluation provide the necessary assurance that the suppression pool temperature response is being adequately calculated for the given inputs. The suppression pool temperature response to a LOCA is primarily input-driven. Therefore, the methods and models used in the evaluation are adequate to calculate the suppression pool temperature response to a large break LOCA. Because the suppression pool temperature response to a MSLB is similar to that of a large break LOCA, the methods and models are considered adequate for calculating the suppression pool temperature response to a MSLB.

3.0 CASE DEVELOPMENT AND DESCRIPTION

Two matrices of potentially limiting scenarios were developed. The first matrix was developed based on capturing the peak post-accident suppression pool temperature. The second matrix was developed from the output of an independent and parallel effort undertaken during the ECCS suction strainer replacement project. The two matrices provide a diverse set of potential accident scenarios for examination. Using these matrices and sensitivity evaluations the scenarios which can result in the highest peak suppression pool temperature are evaluated. From these case matrices, the RELAP and GOTHIC inputs were developed and used to calculate the maximum long-term suppression pool temperature response. Other scenarios were included to ensure that the maximum short-term suppression pools were also calculated.

3.1 Major Sensitivity Parameters

A set of sensitivity parameters was identified to assess the effects on the mass and energy release rates, heat removal from containment, and the long-term mass and energy distribution within the primary containment. These sensitivity parameters provided input to the basis for development of the case matrix.

3.1.1 Offsite Power

The availability of offsite power impacts the mass and energy release. With offsite power available the reactor feed pumps remain powered allowing the operators to continue injection of feedwater during the large break LOCA. Additionally, the availability of offsite power may affect the availability of mitigating equipment after an event.

Feedwater addition may be either conservative or non-conservative with respect to peak torus temperature depending on the temperature of the incoming feedwater. Any feedwater addition at a temperature above the peak suppression pool temperature would be expected to increase peak suppression pool temperature. Conversely, any feedwater addition at a temperature below the peak suppression pool temperature would be expected to decrease peak suppression pool temperature. Only conservative feedwater addition is considered.

Feed addition is considered using two methods. The first method calculates feed injection non-mechanistically based on the addition of all hot feedwater initially contained in the feed piping and considering the structural (piping and component metal) energy contributions. This is referred to as "non-mechanistic feedwater". The second approach is to calculate feedwater addition mechanistically using a detailed feed train model coupled with the vessel model. This is referred to as "mechanistic feedwater".

Availability of mitigating equipment will be determined based on the availability of offsite power and the single failure assumptions made for a given scenario. The availability of mitigating equipment can have a significant impact on the results due to the potential for varying the heat removal from the reactor vessel and subsequent transfer to the wetwell.

3.1.2 Single Active Failure

The suppression pool temperature response is a function of the initial pool temperature and the subsequent rate of heat addition from flow through the break, heat addition from the operating ECCS pumps, and heat removal via the RHR heat exchangers. Single active failure assumptions can change the number of operating RHR or CS pumps affecting the pump heat rate and rate of ECCS injection into the vessel. The rate of ECCS injection controls the rate of energy removal from the vessel and, therefore, energy addition to the suppression pool. Single active failure assumptions can change the number of RHR heat exchangers in service or the flow through the heat exchangers and, therefore, the energy removal from the suppression pool. The results of the Vermont Yankee LOCA single active failure analysis was used in assessing potentially limiting failures.

3.1.3 Post-Accident Containment Cooling Mode

There are three significant potential RHR alignments allowed by procedure that can be used post-accident to cool the containment. The first two are torus cooling and containment spray[2] where at least one loop of the RHR system is realigned, after adequate core cooling is achieved, to take a suction on the suppression pool and discharge through the RHR heat exchanger back to the suppression pool or to the drywell and/or wetwell spray headers. The other method, injection mode cooling, recognizes that plant operating procedures instruct the operators to direct RHRSW flow through the RHR heat exchanger as soon as practicable after LPCI initiates and emergency

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2 Containment spray initiation is considered "torus cooling" for the purposes of this discussion. The difference in the heat transfer phenomena from the reactor vessel to the wetwell is not sufficiently different from torus cooling to warrant separate discussion in this section.

operating procedures may require continued vessel injection with all available ECCS. These two different RHR cooling modes will result in different post-LOCA suppression pool temperatures due to the differences in the rate of energy removal from the vessel and, therefore, energy addition to the suppression pool.

Emergency operating procedures call for the operators to "operate all available torus cooling using only those RHR pumps not required for adequate core cooling". One definition of "adequate core cooling" is one core spray pump injecting at design flow. However, if the operators are basing a judgement of "adequate core cooling" on vessel level alone and are not convinced that adequate core cooling is being achieved, or could be continued with redirection of an RHR pump to torus cooling (given that a large enough break has occurred and that vessel level indication may not be reliable or given that the break is a large suction line break and that safety-related systems alone may not be able to restore vessel level above the top of the core), emergency operating procedures direct them to continue their attempts to flood the vessel. Operators would eventually attempt to initiate containment flood-up if the core cannot be covered. Containment flood-up would add relatively cold water from sources external to primary containment into the primary containment. This cold water addition would most likely arrest any torus heatup and increase the available NPSH by both lowering the torus water temperature and increasing the suppression pool static head. However, containment flood-up may not be guaranteed (assuming a single failure) with safety-related systems. Therefore, this action is not credited in this analysis.

3.1.4 Containment Spray

Spray headers in both the wetwell and the drywell are supplied from the discharge of the RHR pumps after the flow has exited the RHR heat exchanger. These sprays are manually initiated based on guidance provided in the emergency operating procedures. Initiation of containment spray will reduce the wetwell and drywell pressure and temperature due to steam condensation on the cold spray drops and due to the reduction in temperature of the non-condensible gases. The lowest wetwell pressure and air space temperature would be expected based on the coldest spray temperature. The coldest spray temperature is expected if the RHR heat exchanger is operating at its highest efficiency with the coldest service water temperature, maximum RHRSW flow, minimum RHR flow, and no tube plugging in a clean heat exchanger. Additionally, the coldest pool temperature would result in a lower spray temperature. However, NPSH margin may be increased for a cold suppression pool due to the reduction in water vapor pressure.

The competing effects of cold spray and hot suppression pool were not fully evaluated as the NPSH evaluations were not crediting any pressure above atmospheric and only peak suppression pool temperature was required.

3.1.5 RHR Service Water Flow

The RHRSW flow rate impacts the post-LOCA suppression pool temperature and the post-LOCA containment pressure. A decrease in the RHRSW flow will reduce the suppression pool heat removal rate and, thus, increase the post-LOCA peak suppression pool temperature and vice-versa. Therefore, the effects of RHRSW flow and temperature were studied by performing a sensitivity on RHRSW flow (Run 5).

3.1.6 Additional Heat Sources and Sinks

The RHR and CS pumps transmit energy to the fluid in the form of kinetic energy and pressure increase as well as thermal energy from pump inefficiencies. As the single failure assumptions can affect the number of pumps running and therefore the pump energy added to the fluid, the energy added by the pumps is modeled and considered in assessing scenarios to be analyzed. The effects of pump heat due to various single failures is accounted for in the set of runs performed.

Additionally, the water that is initially in the ECCS piping acts as an additional heat sink. Credit is taken for this additional heat sink for those ECCS trains that continue operating long-term.

3.1.7 Additional Uncertainties

Major uncertainties are addressed by analysis to determine their effects on the results. Some specific uncertainties analyzed are:

- closure of the recirculation line discharge valve in the broken loop; a sensitivity is performed assuming the valve remains open due to the valve's potential inability to close under a high differential pressure; however, there is no guarantee that the valve will not close. This uncertainty is studied in Run 2.

- if offsite power is available and an RRU fails, the accompanying room heatup may result in loss of pumps operating in the room with the failed RRU. However, it can be postulated that the pumps do not fail in the time frame of this analysis (<10 hours run time) due to exceptionally robust pumps or cool initial conditions in the room. Therefore, a sensitivity is performed to assess the differences if the operating pumps do not fail (Run 8).

These major uncertainties and parametric variations illustrate the dependence of the results on both ECCS, RHRSW and feedwater system operation. The next section will discuss in more detail how the uncertainty in offsite power and limiting single active failures will lead to the actual cases being examined.

3.2 Limiting ECCS Operation Cases

The purpose of this section is to identify a set of limiting ECCS cases for this analysis. Two sets of cases with varying system operations are developed for two of the three calculation objectives related to determining the long-term containment response. The two resulting sets of system operation cases are independent of break location or type. In other words, given the offsite power availability and single failure assumptions, the system mode of operation is defined. For each system mode of operation, a large recirculation line break or MSLB can be analyzed.

The Vermont Yankee LOCA single failure analysis was examined for single failures consistent with maximizing post-LOCA suppression pool temperature. The single failures of interest are documented in Table 1 along with the effects on the ECCS and the probable effects on the feed system. In support of the replacement ECCS suction strainer bid specification a set of evaluation cases was developed and is also contained in Table 1.

Not all of the cases identified in Table 1 required a full analysis. Many cases were concluded to be non-limiting based on the results of the sensitivity analyses.

The criteria used to choose the cases identified in Table 1 are based on a variety of single failures resulting in different ECCS and feedwater system operation combinations. The base scenario for evaluation is the large break LOCA, a double-ended guillotine rupture of the recirculation piping at the pump suction. Loss of offsite power is not assumed apriori.

A cursory feedwater and condensate system failure modes and effects analysis (FMEA) was performed to support determining the potential effects of the ECCS single failures on the feed and condensate system. The cursory FMEA identified that there could be a wide range of post-accident feed and condensate system operation modes resulting in a wide range of potential post-accident feedwater flow rates. Therefore, post-accident feed flow rate was examined as a sensitivity to identify bounding feed flow conditions.

Operator action is credited to establish torus cooling no later than 10 minutes after event initiation. Other operator action is assumed based on the requirements of a specific sensitivity analysis. For instance, the operators may need to throttle the core spray and/or RHR pumps at some point in the accident to ensure adequate NPSH is maintained. For this particular sensitivity case a specific operator action, to throttle the core spray and/or RHR pumps at a certain time, has been assumed.

3.3 Individual Analysis Case Description

This section provides a description of the subset of the possible cases provided in Table 1 that were analyzed. The set of cases as well as the sensitivities analyzed are given in Table 2.

The base case presented here (identified as Run 0) represents a starting point to make comparisons with the current FSAR long-term containment analysis. It is not intended to be the limiting or bounding case. Unlike the current FSAR long-term containment analysis, this base case includes feedwater addition (although non-mechanistically).

The values for various parameters identified in the following case descriptions are based on the inputs and assumptions provided in Section 4.

Run 0 - Base Case

The postulated event analyzed is a DBA-LOCA (DEG break of the largest recirculation piping, nominally on the suction side of the recirculation pump). Coincident with this break is a single-active failure of the DC-1 electrical bus (see Run 1 description). ECCS and RHRSW flows are taken at their nominal values. Non-mechanistic feedwater addition is considered and is terminated when all conservatively hot feedwater has been injected.

Run 1 - DC-1 Single Failure

This case represents a DBA-LOCA with no loss of offsite power (LOOP). The break is assumed to be on the A recirculation loop consistent with LOCA modeling methods (a break in B recirculation loop would be symmetric). The single failure is assumed to be a failure of the DC-1 bus. Equipment availability is consistent with Table 1, Peak Suppression Pool Temperature Case 2. ECCS flows are assumed to be at their minimum values.

RHR System Short Term Operation - two RHR pumps are available for LPCI mode of operation, one in each loop; the other two RHR pumps are not available as a result of the DC-1 failure; the flow from the LPCI pump aligned to the broken loop is injected into the vessel as the recirc discharge valve is assumed to shut; the other pump injects into the vessel through the intact recirculation loop.

RHR System Long Term Operation - at 10 minutes the operators have aligned the RHR system in the intact loop to torus cooling. In the realignment the RHR heat exchanger bypass is closed and the RHRSW pump is started. The broken loop RHR pump fails due to failure of the associated RRU.

CS System Operation - one core spray pump is available and injects water into the vessel for the duration of the transient; the other core spray pump is unavailable due to the DC-1 failure. The operator is assumed to throttle the core spray pump to design flow at 10 minutes.

Feed & Condensate System Operation - offsite power is available and only two condensate pumps (no feedpumps) are available for feedwater injection due to the DC-1 failure; the operators continue to allow feedwater injection through the condensate pumps as one of the preferred water sources; feedwater injection begins when vessel pressure is reduced sufficiently below the head capacity of the condensate pumps (for modeling purposes, feedwater is assumed to continue injecting immediately after the break occurs as the vessel pressure is reduced almost immediately); feedwater injection terminates when all conservatively hot feedwater has been injected.

Run 2 - Recirc Discharge Line Valve Sensitivity

This case provides an analysis of the sensitivity of the results to broken loop recirc discharge line valve failure. Run 1 is analyzed but instead of the broken loop recirc discharge line closing, the valve is assumed to fail open. Therefore, the Run 1 RHR system operation is changed such that any injection into the broken loop is spilled out the broken loop to the drywell instead of injecting to the vessel.

Run 3 - RHR Heat Exchanger Failure

This case represents a DBA-LOCA with no LOOP. The break is assumed to be on the A recirculation loop consistent with LOCA modeling methods (a break in B recirculation loop would be symmetric). The single failure is assumed to be a failure that prevents the heat removal capability of one RHR heat exchanger (e.g. MOV-89 A/B fails shut). Equipment availability is consistent with Table 1, Peak Suppression Pool Temperature Case 1. ECCS flows are assumed to be at their minimum values.

RHR System Short Term Operation - all four RHR pumps are available for LPCI mode of operation; the flow from the two LPCI pumps aligned to the broken loop is injected into the vessel due to the closure of the recirc discharge valve; the other two pumps inject into the vessel through the intact recirculation loop.

RHR System Long Term Operation - at 10 minutes the operators are assumed to have realigned the RHR train in the broken loop for torus cooling. In the realignment one RHR pump is secured, the RHR heat exchanger bypass is closed and the associated RHRSW pump is started. The intact loop RHR heat exchanger is inoperable as a result of a single active failure. The operators continue to allow the remaining two RHR pumps to inject into the RPV, as allowed by the emergency operating procedures, for additional core cooling. In addition, the operators are assumed to have throttled the RHR pumps to design flow.

CS System Operation - two core spray pumps are available and inject water into the vessel for the duration of the transient. At 10 minutes, the operators are assumed to have throttled the pumps to design flow.

Feed & Condensate System Operation - offsite power is available as are the two running feedwater pumps and three condensate pumps; the operators continue to allow feedwater injection as one of the preferred water sources; feedwater injection terminates when all conservatively hot feedwater has been injected.

Run 4 - ECCS Flow Sensitivity

This case provides a sensitivity of the results to ECCS flow rate. Run 1 is analyzed but instead of the ECCS pumps delivering the minimum expected flows, the pumps are assumed to deliver the maximum expected flows. An exception to this is the flow through the RHR heat exchanger that is actively removing heat as the operator sets the RHR flow through the heat exchanger using procedural guidance. This limits the sensitivity to investigating the effects of the increased flushing of heat from the vessel and drywell to the suppression pool due to the increased ECCS flow.

Run 5 - RHRSW Flow Sensitivity

This case provides a sensitivity of the results to RHRSW flow rate. Run 4 is analyzed but, instead of using the minimum RHRSW flow, an increased RHRSW flow is assumed to obtain the sensitivity. This increases the rate of heat removal through the RHR heat exchanger.

Run 6 - Short-Term Torus Temperature Sensitivity

This case provides an examination of the highest suppression pool temperature early (approximately < 1800 sec) in the event. It is anticipated that the operators will throttle core spray pump (and possibly RHR pump) flow at some time after a large break LOCA has occurred. This action will assist in preserving (or re-gaining) NPSH margin. In order to provide additional information to assist in determining the minimum required time for pump throttiing this sensitivity uses inputs which would tend to result in a higher suppression pool temperature early in the event. These inputs may not be the same as those chosen to maximize the peak suppression pool temperature.

A RHR heat exchanger failure similar to Run 3 is analyzed with parametric inputs selected to maximize early heat addition. These include the assumption that the recirc discharge valve in the broken loop shuts normally and ECCS flows are at their maximum values. In addition, no throttling of the injecting ECCS pumps is assumed so as to provide an assessment of the length of time the operators have available to throttle the ECCS pumps.

Run 7 - Feed Rate Sensitivity

This case determines the sensitivity of the torus temperature response to the feed rate during the peirod of feedwater addition. The single failures that have been examined may have an effect on the feed and condensate pump availability. Since the feed rate can affect the rate at which hot water is flushed from the vessel to the suppression pool this sensitivity seeks to assess the dependence of the results on feed rate. Based on review of the feed and condensate pump head vs flow curves and design point, it is postulated that the flow rate through the feed system could be double the normal feed rate. Therefore, double the normal feed rate is arbitrarily assumed for the sensitivity.

Run 8 - RHR Pump Failure Sensitivity

This case determines the sensitivity of the suppression pool temperature response to RHR pump failure resulting from a loss of room cooling following a DC-1 or DC-2 failure. Run 4 postulates that the RHR pump operating in the room with the failed RRU (due to the DC-1 single failure) ceases to operate at 10 minutes post-LOCA based on the Vermont Yankee LOCA single failure analysis. However, it is not guaranteed that the RHR pump will fail as the room may be cooler than initially assumed or the pump may be more robust than assumed in the single failure analysis. Given the energy an RHR pump can potentially add to the containment (in the form of thermal energy due to pump losses or conversion of the pump's work to thermal energy due to system friction and form losses), it is not clear that the assumption of RHR pump failure due to room heatup is conservative. Therefore, this sensitivity assumes the RHR pump continues to operate. Run 4 includes consideration of maximum ECCS flow rates which was determined to be limiting with respect to a DC-1 or DC-2 failure.

4.0 ASSUMPTIONS, INPUTS AND INITIAL CONDITIONS

Initial conditions and plant parameters used as input into the analysis maintain the overall conservatism by maximizing the suppression pool temperature. Where appropriate, instrument uncertainties have been explicitly considered in developing the analysis inputs. Table 3 provides values of key input parameters.

Various assumptions are made during the course of the analysis. Some have been explicitly identified where they are most appropriate. Other key assumptions used throughout this analysis are identified here:

1. Operator action is taken to place one RHR loop in torus cooling by 10 minutes. No operator action, other than diagnosis, verification and procedure initiation, is credited prior to 10 minutes after the accident. In other words, the operators are assumed to enter the procedures for starting torus cooling prior to ten minutes but torus cooling is not considered initiated until 10 minutes. This is consistent with the current containment analysis design and licensing basis for crediting operator actions.

2. Operator action is taken to throttle core spray pump flow to ensure long-term pump flow is near design flow (3000 to 4000 gpm; 3000 gpm based on minimum required for adequate core spray coverage, 4000 gpm is based on maximum flow allowed to assure adequate NPSH). The time that this throttling occurs is assumed to be 10 minutes.

3. RCIC and HPCI are not available. RCIC and HPCI require steam from the vessel to power the turbine-driven pump. Since the accidents analyzed here rapidly depressurize the vessel, there is insufficient steam to assume that RCIC and/or HPCI operate.

5.0 RESULTS

The ten minute and peak suppression pool temperature results are provided in Table 4, for the DC-1 single failure case and its sensitivities, and Table 5, for the RHR heat exchanger failure case and its sensitivities. The tables also provide a summary descriptions of the cases analyzed with key equipment operability parameters as well as a summary description of the ECCS flows. The cases are divided according to their basic single failure assumption. The results from the current FSAR suppression pool temperature analysis is provided for comparison.

The bounding combination of DBA-LOCA sensitivities, as well as other events (small break LOCA, steam line breaks, and safe shutdown events), have been assessed. The conservative assessment concluded that the peak suppression pool temperature following all events was limited to less than 185°F. For the limiting DBA-LOCA the maximum ten minute suppression pool temperature was assessed to be less than 164°F and the peak suppression pool temperature was conservatively assessed to be 182.6°F. This peak suppression pool temperature includes an allowance for injection mode cooling which was not specifically modelled in Runs 0 through 8 of Tables 4 and 5.

6.0 REFERENCES

1 R. T. Fernandez, et. al., "RELAP5YA - A Computer Program for Light-Water Reactor System Thermal Hydraulic Analysis," YAEC-1300P, October 1982, Revised June 1993

2 GOTHIC Containment Analysis Package, Version 5.0e, Users Manual, NAI 8907-02 Rev. 6

Table 1 - Equipment Availability Matrix

Peak Suppression Pool Temperature Cases

Case	Single Failure	RHR-A (Recirc Loop B)	RHR-B (Recirc Loop A)	CS	Feed & Condensate	Comments
1	RHR HX	ST - 2 pumps LPCI LT - 2 pump LPCI	ST - 2 pumps LPCI LT - 2 pump SPC or IMC	ST - 2 pumps LT - 2 pumps	3 condensate & 2 feed pumps	e.g. MOV-89 A/B fails closed
2	DC-1	ST - 1 pump LPCI LT - 1 pump SPC or IMC	ST - 1 pump LPCI LT - 0 pumps	ST - 1 pump LT - 1 pump	2 condensate pumps	ECCS Train B fails LT - RHR pump fails due to RRU failure
3	DC-2	ST - 1 pump LPCI LT - 0 pump LPCI	ST - 1 pump LPCI LT - 1 pump SPC or IMC	ST - 1 pump LT - 1 pump	1 condensate & 2 feed pumps	ECCS Train A fails LT - RHR pump fails due to RRU failure
4	RRU 7	ST - 2 pumps LPCI LT - 0 pumps	ST - 2 pumps LPCI LT - 2 pump SPC or IMC	ST - 2 pumps LT - 1 pump	3 condensate & 2 feed pumps	

ECCS Suction Strainer Replacement Cases

Case	Single Failure	RHR-A (Recirc Loop B)	RHR-B (Recirc Loop A)	CS	Feed & Condensate	Comments
0	DC-2 (w/ LOOP)	ST - 1 pump LPCI LT - 0 pumps	ST - 1 pump LPCI LT - 1 pump SPC or IMC	ST - 1 pump LT - 1 pump	Flashing or draining only	LT - RHR-A pump fails due to RRU 7 failure
1	None	ST - 2 pumps LPCI LT - 2 pumps LPCI	ST - 2 pumps LPCI LT - 2 pump SPC or IMC	ST - 2 pumps LT - 2 pumps	3 condensate & 2 feed pumps	Operators continue injection with RHR-A
2/3	UPS-1A/B	ST - 2 pumps LPCI LT - 2 pumps LPCI	ST - 2 pumps min flow LT - 1 pump SPC	ST - 2 pumps LT - 2 pumps	3 condensate & 2 feed pumps	MOV-27A/B fails closed MOV-53A/B fails open
4	1 CS Pump	ST - 2 pumps LPCI LT - 2 pumps LPCI	ST - 2 pumps LPCI LT - 2 pump SPC or IMC	ST - 1 pump LT - 1 pump	3 condensate & 2 feed pumps	
5/6	1 RHR Pump	ST - 2 pumps LPCI LT - 2 pumps SPC or IMC	ST - 1 pump LPCI LT - 1 pump SPC or IMC	ST - 2 pumps LT - 2 pumps	3 condensate & 2 feed pumps	
7/8	RRU 7	ST - 2 pumps LPCI LT - 0 pumps	ST - 2 pumps LPCI LT - 2 pumps SPC or IMC	ST - 2 pumps LT - 1 pump	3 condensate & 2 feed pumps	

Notes:
(1) ST = Short Term (< 600 seconds), LT = Long Term (> 600 seconds)
(2) LPCI = pump operating in LPCI mode; SPC = pump operating in suppression pool cooling mode; IMC = pump operating in injection mode cooling where the pump is operating in LPCI mode but with RHRSW also operating allowing heat removal through the RHR heat exchanger. IMC is not credited in this analysis. It is identified here only for the sake of completeness. It is also assumed that only one RHR pump is used when an RHR train is placed in torus cooling. However, the idle pump may be available if otherwise needed.
(3) For those cases with RRU failure (explicit or due to DC-1 or DC-2 failure) combinations of failed pumps in room with failed cooler is allowed.

Table 2 - Index of Analysis Cases

Run	Title	Description[3]
0	Base Case	LBLOCA with DC-1 failure, normal power available, ECCS flows at nominal values, non-mechanistic feed
1	DC-1 Failure	Table 1, Peak Suppression Pool Temperature Case 2, LBLOCA with DC-1 failure and normal power available, ECCS flows at minimum values, broken loop recirc discharge line valve shuts
2	Recirc Discharge Line Valve Sensitivity	Run 1 but the broken loop recirc discharge line valve remains open
3	RHR Heat Exchanger Failure	Table 1, Peak Suppression Pool Temperature Case 1, LBLOCA with failure of MOV-89A and normal power available, ECCS flows at minimum values; recirc discharge line valve shuts
4	ECCS Flow Sensitivity	Run 2 but the ECCS pumps are running at maximum expected flow when injecting into the RPV
5	RHRSW Flow Sensitivity	Run 4 but the RHRSW flow is at 2850 gpm
6	Short-Term Torus Temperature Sensitivity	Run 3 but with inputs selected to maximize the short-term (~10 minutes) suppression pool temperature
7	Feed Rate Sensitivity	Run 3 but with feed rate increased
8	RHR pump failure sensitivity	Run 4 but the operating RHR pump in the room with the inoperable RRU does <u>not</u> fail due to room heat up

[3] Unless otherwise stated:
- the feedwater addition is assumed to be mechanistically determined
- the recirc discharge valves in both loops are assumed to close normally
- torus cooling is assumed to be initiated at 10 minutes
- core spray pumps are assumed to be throttled to minimum design flow at 10 minutes

Additional information is contained in the detailed case descriptions.

Table 3 - Initial Conditions and Input Values

Parameter	Analysis Value	Comments
Initial Reactor Power	1625 MWth	100% power + 2% calorimetric uncertainty[4]
Core Decay Heat	ANS 5.1 +2σ	ANS 5.1 1979 standard + 2σ uncertainty[4]
MSIV Closure Time	3.0 sec	Tech. Spec. Minimum value (turbine stop valve fast closure is considered)
Core Spray Flow	<600 sec - 3708 - 4600 gpm @ 0 psid > 600 sec - 3000-4000 gpm	
RHR Pump Flow (< 600 sec)	Single pump in train: 5640 - 7400 gpm @ 0 psid Two pumps in train: 11104 - 14200 gpm @ 0 psid	
RHR Flow (>600 sec)	Single pump in train: 6400 - 7000 gpm Two pumps in train: 11104 - 14200 gpm	
RHR Hx Tube Plugging	5%	
RHR Hx Fouling	0.0005 shell, 0.002 tube	Heat exchanger design condition
RHRSW Flow	2700 gpm	
Wetwell Temperature	90°F	Proposed Technical Specification Limit
Wetwell Water Volume	68,000 ft^3	Minimum Technical Specification Limit
RHRSW Inlet Temperature	85°F	Maximum Service Water System design condition

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4 Reactor power uncertainty and decay heat uncertainty are statistically combined.

Table 4 - Summary of Torus Temperature Results - DC-1 Single Failure

Case	Title	ECCS Flows		Torus Water Temperature		
		Short Term (0-10 min)	Long Term (> 10 min)	Short Term (10 min)	Long Term Peak	Time of Peak
	FSAR Section 14.6.3 Long-Term Analysis	N/A	1 RHR @ 7000 gpm	140°F	166°F	~17000 sec
Run0	Base Case	1 CS @ 4600 gpm 1 LPCI @ 7400 gpm	1 CS @ 3000 gpm 1 RHR @ 7000 gpm	154.5°F	173.3°F	18695 sec
Run1	DC-1 Single Failure	1 CS @ 3708 gpm 2 LPCI @ 5640 gpm ea.	1 CS @ 3000 gpm 1 RHR @ 6400 gpm	159.8°F	175.7°F	18105 sec
Run2	Recirculation Discharge Valve Sensitivity (broken loop recirc discharge valve fails open)	1 CS @ 3708 gpm 1 LPCI @ 5640 gpm	1 CS @ 3000 gpm 1 RHR @ 6400 gpm	158.6°F	175.7°F	17930 sec
Run4	ECCS Flow Sensitivity (Run 1 but max ECCS injection flow)	1 CS @ 4600 gpm 2 LPCI @ 7400 gpm	1 CS @ 4000 gpm 1 RHR @ 6400 gpm	161.0°F	176.2°F	17200 sec
Run5	RHRSW Flow Sensitivity (Run 4 but RHRSW flow is 2850 gpm)	1 CS @ 4600 gpm 2 LPCI @ 7400 gpm	1 CS @ 4000 gpm 1 RHR @ 6400 gpm	161.0°F	175.5°F	16585 sec
Run8	RHR Pump Failure Sensitivity (Run 4 but RHR pump in room with failed RRU continues to operate)	1 CS @ 4600 gpm 2 LPCI @ 7400 gpm	1 CS & 1 LPCI @ 11000 gpm total 1 RHR @ 6400 gpm	161.0°F	178.8°F	17275 sec

Table 5 - Summary of Torus Temperature Results - RHR Heat Exchanger Single Failure

Case	Title	ECCS Flows		Torus Water Temperature		
		Short Term (0-10 min)	Long Term (> 10 min)	Short Term (10 min)	Long Term Peak	Time of Peak
Run3	RHR Heat Exchanger Failure	2 CS @ 3708 gpm ea. 4 LPCI @ 11104 gpm each train	2 CS & 2 LPCI @ 18800 gpm total 1 RHR @ 6400 gpm	161.7°F	181.2°F	19735 sec
Run6	Short Term Temperature Sensitivity (Run 3 but with max. ECCS injection flow)	2 CS @ 4600 gpm ea. 4 LPCI @ 14200 gpm each train	2 CS & 2 LPCI @ 23400 gpm total 1 RHR @ 6400 gpm	162.2°F	181.3°F	19634 sec
Run7	Feed Flow Sensitivity (Run 3 but with double feed flow rate)	2 CS @ 3708 gpm ea. 4 LPCI @ 11104 gpm each train	2 CS & 2 LPCI @ 18800 gpm total 1 RHR @ 6400 gpm	162.4°F	181.3°F	19528 sec

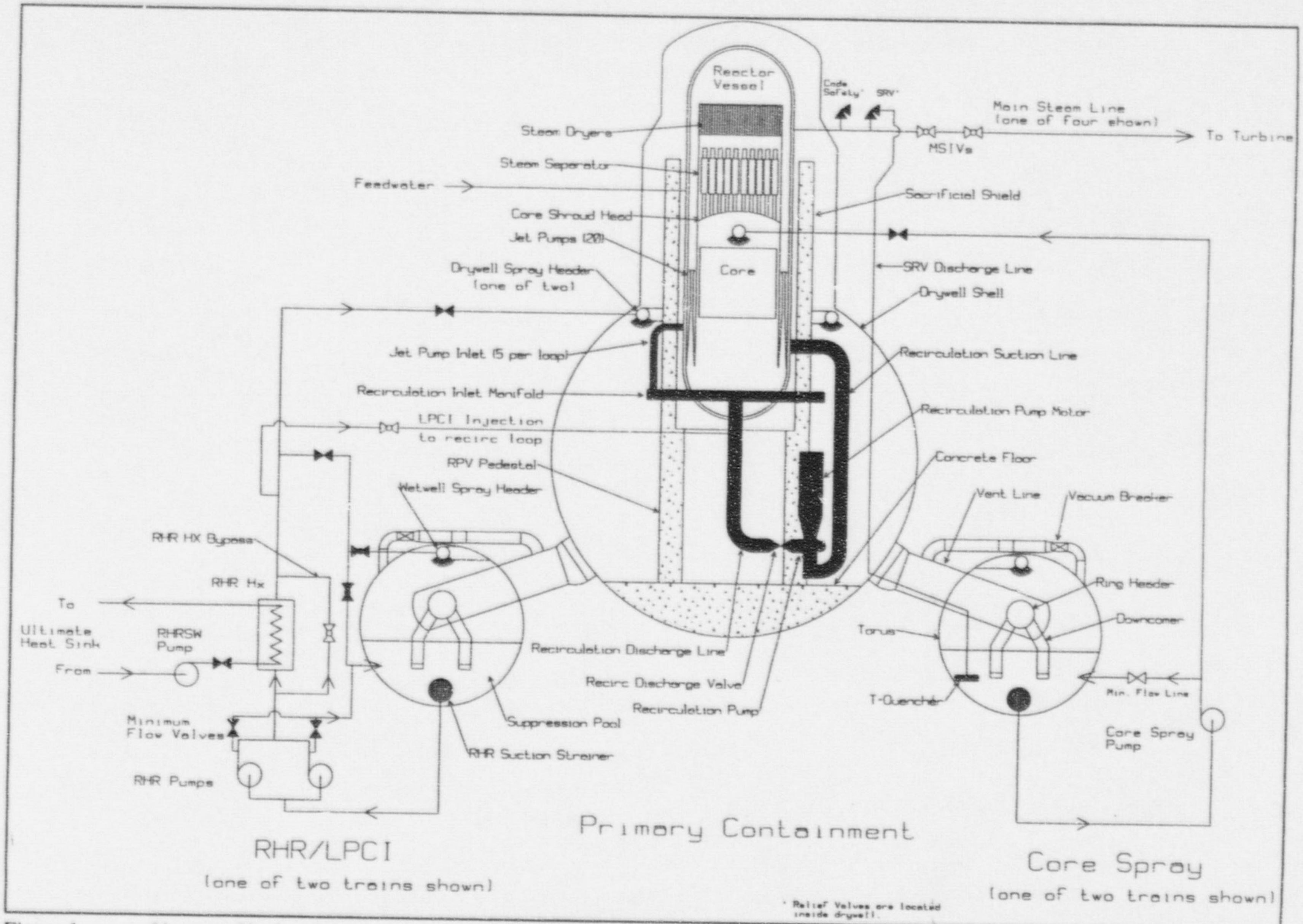

Figure 1 Vermont Yankee Primary Containment and ECC System Schematic