15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

The results of event analyses for the current cycles are in Appendix 15C for SSES Unit 1 and Appendix 15D for SSES Unit 2. Appendix 15E contains information and analytical results that are for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

15.1.1 LOSS OF FEEDWATER HEATING

FANP NRC approved methods (Reference 15.1-2) have identified this event as a limiting event, and therefore, it has been evaluated for the current cycles for Units 1 and 2.

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

A feedwater heater can be lost in two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater (Although applicable to some BWRs, this capability does not apply to Susquehanna).

The first case produces a relatively gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of that feedwater occurs. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. This event has been conservatively estimated to incur a loss of up to 100°F of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. The event can occur with the reactor in either the automatic or manual control mode. In automatic control, some compensation of core power is realized by modulation of core flow, so the event is less severe than in manual control. However, the automatic flow control mode has been disabled for the SSES Units. For this reason, the loss of feedwater heating for current cycles has been analyzed only for the reactor in the manual control mode.

15.1.1.1.2 Frequency Classification

The probability of this event is considered low enough to be categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency.

This event is analyzed under worst case conditions of a 100°F loss and full power although a reduction of feedwater temperature of 100°F at high power has never been reported.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Tables 15C.1.1-1 and 15D.1.1-1 list the sequence of events for this transient for the reactor in the manual control mode for Units 1 and 2.

15.1.1.2.1.1 Identification of Operator Actions

If no automatic recirculation runback occurs, the reactor operator is to reduce power according to off-normal reactor operating procedures. The operator monitors the core for instabilities and monitors operating conditions versus the Power/Flow map. The operator then examines the operation of the feedwater heaters and takes necessary corrective action and resumes normal operation. If reactor scram occurs, although it is not predicted, the operator is directed to perform those actions listed in Table 15E.1.1-1.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that the plant instrumentation and controls, plant protection and reactor protection systems functioned normally.

The average power range monitor (APRM) provides the alarm to the operator, but no protection system trip is expected or required to mitigate the predicted consequences of this event.

Required operation of Engineered Safeguard Features is not expected for either of these transients.

15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The APRM alarm alerts the operator, however, the reactor requires no automatic trip. Therefore, single failures are not expected to result in a more severe event than analyzed. See Appendix 15A.

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The loss of feedwater heating is a relatively slow transient and has been conservatively analyzed by determining the final steady-state reactor operating condition assuming no operator or control system action were to occur. The analysis method is described in Reference 15.1-2. It has been shown that the method conservatively bounds the current fuel assembly designs.

Startup test data for the Susquehanna reactors show that a loss of feedwater heaters results in a drop in feedwater temperature on the order of 40°F, (Reference 15.1-3) which is considerably less than the 100°F drop in temperature assumed for the analysis performed here.

Therefore the analysis results reported herein will bound the expected transient conditions if this anticipated operational occurrence were to occur.

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2 and 15D.0-2 for the current cycles for Units 1 and 2.

The plant was analyzed at power levels of 100% and 50%. For the 100% power condition, a range of flows were analyzed. For the 50% power condition, the flow was assumed to be 62 Mlbs/hr.

Since this transient is relatively slow it can be conservatively analyzed by determining the change in the core thermal conditions based on initial and final steady-state conditions.

The transient is simulated by running an initial condition steady-state 3-D calculation and determining the MCPR and the corresponding eigenvalue for this state-point. The calculation is then repeated. The xenon distribution is kept the same as that determined for the initial state-point. The sub-cooling is changed to correspond to the value that would be obtained if the feedwater temperature decreased 100°F. The reactor power is increased until the eigenvalue matches the initial steady-state value. The MCPR is determined for this second condition and the ΔCPR is determined by taking the difference between this MCPR and the MCPR from the initial state-point.

15.1.1.3.3 Results

In manual mode, no compensation is provided by core flow and thus the power continues to increase. Vessel steam flow increases and the initial system pressure increase is small. The increased core inlet subcooling aids core thermal margins. ΔCPR's determined for the current cycles for Units 1 and 2 are given in tables 15C.0-1 and 15D.0-1.

The maximum steady-state power for this event is less than the 122% high neutron flux analytical setpoint. It was also determined that for the current cycles for this event, the LHGR power transient limit was not violated.

The thermal margin was also determined for reduced powers for two loop operation and single loop operation. The results indicate that the ΔCPRs for two loop operation are more limiting than for single loop operation and therefore can be used to establish the MCPR operating limits for single loop operation for this event.

15.1.1.3.4 Considerations of Uncertainties

The magnitude of the feedwater temperature change assumed was more severe than what is believed possible for the Susquehanna Units. Since the analysis conservatively assumed steady-state conditions at the end of the transient, realistic physics and thermal-hydraulic parameters were used.

15.1.1.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 Radiological Consequences

Since this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment there are no radiological consequences associated with this event.

15.1.2 FEEDWATER CONTROLLER FAILURE - MAXIMUM DEMAND

This event has been identified as a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

15.1.2.1.2 Frequency Classification

This event is an incident of moderate frequency.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

With excess feedwater flow the water level rises to the high-level reference point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Tables 15C.1.2-1 and 15D.1.2-1 list the sequence of events for this transient for Units 1 and 2. Figures 15C.1.2-1 and 15D.1.2-1 show the changes in important variables during this transient for the current cycles for Units 1 and 2.

15.1.2.2.1.1 Identification of Operator Actions

(1) Observe that the high level feedwater pump trip has terminated the failure event.

(2) Switch the feedwater controller from auto to manual control in order to try to regain a correct output signal.

(3) Identify causes of the failure and report all key plant parameters during the event.

15.1.2.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high water level tripping of the main turbine, turbine stop valve scram trip initiation, recirculation pump trip (RPT), and low water level initiation of the Reactor Core Isolation Cooling system (RCIC) and the High Pressure Coolant Injection system (HPCI) to maintain long-term water level control following the trip of the feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

In Tables 15C.1.2-1 and 15D.1.2-1, the first sensed event to initiate corrective action to the transient is the vessel high water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 set point. At this point in the logic a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature."

However, high moisture levels entering the turbine will be detected by high levels in the turbine's moisture separators which results in a trip of the unit.

Scram trip signals from the turbine are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. See Appendix 15A for further discussion.

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of the SSES Units 1 and 2. The methods for modeling and analyzing this event for Unit 1 are described in References 15.1-5 and 15.1-6. The methods for modeling and analyzing this event for Unit 2 are described in References 15.1-4 and 15.1-5.

The nonlinear computer simulated analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

a. An integrated one-dimensional core model is assumed which includes a detailed description of hydraulic feedback effects, axial power shape changes, and reactivity feedbacks.

b. The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core.

c. The steam lines are modeled using pressure nodes incorporating mass and momentum balances which will predict any wave phenomena present in the steam line during pressurization transient.

d. The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

e. Principal controller functions such as feedwater flow, recirculation flow, reactor water level, and pressure, are represented together with their dominant nonlinear characteristics.

f. The ability to simulate necessary reactor protection system functions is provided.

g. The control systems and reactor protection system are modeled.

15.1.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15C.0-2 and 15D.0-2 for the current cycles for Units 1 and 2.

The transient model for the SSES Units 1 and 2 was initialized and executed for this event at one or more exposure steps for the current cycles. The initialization includes both the physics and thermal-hydraulic input that is exposure dependent. The Feedwater Controller Failure is analyzed for each of these exposures to determine the most limiting conditions for the cycles. The analyses are also performed over a range of power levels for Unit 1 and Unit 2. The flow was held constant at 108 Mlbs/hr. In general, the limiting initial condition for this event is full flow of 108 Mlbs/hr. If there is reason to believe that the limiting initial flow condition is other than full flow, additional analyses are performed at lower flows.

The analyses also consider the following:

1. Steam bypass and Recirculation Pump Trip operable,

2. Steam bypass inoperable and Recirculation Pump Trip operable,

3. Steam bypass operable and Recirculation Pump Trip inoperable.

4. Realistic Scram Insertion Time and Maximum Allowable Scram Insertion Time.

The analysis is performed using relief/safety valve setpoints corresponding to the “safety mode.”

The initiating event for this transient is the failure of the feedwater control system causing a step change of feedwater flow from its initial steady-state value to the maximum value of feedwater flow for the analyzed configuration.

The results of these analyses are used to establish the MCPR operating limits as a function of power. These analyses are performed prior to the start of each cycle for SSES Units 1 and 2.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown in Figures 15C.1.2-1 and 15D.1.2-1 for the current cycles of SSES Units 1 and 2. The high water level turbine trip and feedwater pump trip are initiated at the times shown in Tables 15C.1.2-1 and 15D.1.2-1 for Units 1 and 2. Scram occurs simultaneously from stop valve closure, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. If the turbine bypass system is operable, it opens to limit peak pressure in the steam line and the reactor pressure vessel. Peak pressures are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2. These pressures are well below the design limit of 1375 psig. Therefore the nuclear system process barrier pressure limit is not endangered.

The ΔCPRs for this transient are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2.

The bypass valves subsequently close to re-establish pressure control in the vessel during shutdown. The water level will gradually drop to the Low Level isolation reference point, activating the RCIC/HPCI systems for long term level control.

15.1.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable response ((e.g., relief setpoints, scram stroke time (realistic and maximum allowable), and analytical setpoints for reactor protection system trips)). Plant behavior is, therefore, expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual, therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.3 PRESSURE REGULATOR FAILURE - OPEN

This event has been identified as non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for power uprate conditions and the results are applicable to the current cycles for Units 1 and 2.

15.1.3.1 Identification of Causes and Frequency Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow to approximately 129% of rated flow. The maximum steam flow controller setting is 125% of the maximum flow through the turbine control valves, which is 103%, (turbine control valves fully open). Therefore failure of the controller to its maximum results in steam flow demand of 129% of rated flow. The percent flows listed above are based on a rated power of 3441 MWt.

If either the controlling pressure regulator or the backup regulator fails to the open position, the turbine control valves can be fully opened and the turbine bypass valves can be opened until the maximum steam flow is established.

15.1.3.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

Tables 15C.1.3-1 and 15D.1.3-1 list the sequence of events for this transient for Units 1 and 2. Figures 15C.1.3-1 and 15D.1.3-1 show the transient behavior of important system parameters for Units 1 and 2.

15.1.3.2.1.1 Identification of Operator Actions

When regulator trouble is preceded by spurious or erratic behavior of the controlling device, it may be possible for the operator to transfer operation to the backup controller in time to prevent the full transient. If the reactor scrams as a result of the isolation caused by low pressure sensed prior to the main turbine inlet in the run mode, the following is the sequence of operator actions expected during the course of the event. Once isolation occurs the pressure will increase to a point where the relief valves open. The operator should perform those scram actions indicated in Table 15E.1.1-1, and enter the appropriate Emergency Operating Procedure.

Prior to reactor restart, the operator should complete the scram report and initiate a maintenance work authorization of the pressure regulator.

15.1.3.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems except as described below.

Initiation of HPCI and RCIC system functions will occur when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before effects are realized. If these events occur, they will follow some time after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.1.3.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a loss of pressure control such that the increased steam flow demand causes a depressurization. Instrumentation for pressure sensing of the turbine inlet pressure is designed to be single failure proof for initiation of MSIV closure.

Reactor scram sensing, originating from limit switches on the main steam line isolation valves, is designed to be single failure proof. It is therefore concluded that the basic phenomenon of pressure decay is adequately terminated. See Appendix 15A for a further discussion.

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The nonlinear dynamic model described in Reference 15.1-3 was used to simulate this event. This event was re-evaluated for U2C21 utilizing Reference 15.1-4 and confirmed to be non-limiting.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the controlling regulator output to a high value, which causes the turbine control valves to open fully. The model initiates the opening of the bypass valves by a trip signal (at the time the controller fails). The valve then opens based on an input table of valve position versus time. Since the controlling and backup regulator outputs are gated by a high value gate, the effect of such a failure in the backup regulator would be exactly the same. A regulator failure with approximately 130% of rated steam flow was simulated as a worst case.

A 5-second isolation valve closure instead of a 3-second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure setpoint for main steam line isolation initiation. This is within the specification limits of the valve and tends to aggravate the results of the analysis.

For purposes of the analysis the MSIV isolation on low main steam line pressure was conservatively set at its analytical value of 825 psig instead of the nominal value of 861 psig.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Tables 15C.0-2 and 15D.0-2.

15.1.3.3.3 Results

Figures 15C.1.3-1 and 15D.1.3-1 show the response of important nuclear system variables for this transient for Units 1 and 2. The turbine inlet pressure decreases to the low pressure trip setpoint and initiates trip of the MSIV's. Closure of the MSIV's initiates scram, and stops subsequent loss of steam to the main turbine and the feedwater turbines.

Reactor low turbine pressure trip limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the nuclear system process barrier. After the rapid portion of the transient is complete the nuclear system safety/relief valves operate intermittently to relieve the pressure rise that results from decay heat generation. No significant reductions in fuel thermal margins occur. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system and because the safety/relief valves operate only to relieve the pressure increase caused by decay heat, the nuclear system process barrier is not threatened by high internal pressure for this pressure regulator malfunction.

The ΔCPR was determined for this event for the power uprate condition and found to be on the order of 0.01 which indicates that this event does not significantly reduce core thermal margins, as noted above, and therefore does not need to be analyzed for each cycle.

15.1.3.3.4 Consideration of Uncertainties

If the maximum flow limiter were set higher or lower than normal, a faster or slower loss in nuclear steam pressure would result. The rate of depressurization may be limited by the bypass capacity.

For example, the turbine control valves will open to the valves wide-open state admitting slightly more than the rated steam flow, and with the limiter in this analysis set at 125%, (130% rated steam flow), we would expect approximately 25% steam flow to be bypassed. This is essentially the maximum bypass flow and a faster rate of depressurization due to a pressure regulator failure is therefore not possible.

Depressurization rate has a proportional effect upon the voiding action of the core. If it is large enough, the sensed vessel water level trip set point (L8) may be reached initiating a turbine and feedwater pump trip early in the transient. Turbine trip will initiate reactor scram and shut down the reactor. Thermal margins will be better than a typical turbine trip event because of the power reduction initially experienced due to increased core voids in this event. Since the pressure regulator failure continues to signal the bypass to remain fully open, the turbine inlet pressure will drop below the low pressure isolation setpoint and the expected transient signature will conclude with an isolation of the main steam lines.

15.1.3.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed. Peak pressure in the bottom of the vessel is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary. Vessel dome pressure is shown in Figures 15C.1.3-1 and 15D.1.3-1 for Units 1 and 2.

15.1.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual, therefore this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.4 INADVERTENT SAFETY/RELIEF VALVE OPENING

This event has been identified as non-limiting, and therefore, it is not explicitly analyzed each cycle.

Inadvertent opening of a safety/relief valve can lead to two possible events. First, the valve may "open" and "re-close". This event has no significant effect on plant operation. Second, the valve may "open" and stick in the "open" position. This is the more limiting case and results in the plant transient discussed below.

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. Opening and closing circuitry at the individual valve level (as opposed to groups of valves) is subject to a single failure impact. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance is categorized as an infrequent incident but due to a lack of a comprehensive data basis, it is being analyzed as an incident of moderate frequency.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-1 lists the sequence of events for this transient.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must "re-close" the valve as soon as possible and check that the reactor and T-G output return to normal.

15.1.4.2.2 Systems Operation

In this transient, the core performance analysis assumes normal functioning of plant instrumentation and controls, specifically the pressure regulator and level control systems. Additionally, normal operation of relief valve discharge line temperature sensors and the suppression pool temperature sensors provides operator information as the basis for initiating a timely plant shutdown.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

Failure of additional components (e.g., pressure regulator, feedwater flow controller) is discussed elsewhere in Chapter 15. In addition, a detailed discussion of such effects is given in Appendix 15A.

15.1.4.3 Core and System Performance

15.1.4.3.1 Mathematical Model

It was determined that this event is not limiting from a core performance standpoint. Therefore a qualitative presentation of results is described below.

15.1.4.3.2 Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level of 3510 Mwt (4032 Mwt was used for dose analyses) when a safety/relief valve is inadvertently opened. Manual recirculation flow control is assumed. Flow through the valve at normal plant operating conditions stated above is approximately 928,800 lbs/hr.

15.1.4.3.3 Qualitative Results

The opening of a safety/relief valve allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient. The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and therefore the safety limit margin is unaffected.

Continued maximum steam flow to the suppression pool will be terminated by operator action.

15.1.4.4 Barrier Performance

As discussed above, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following capability. RCPB and containment design limits are not exceeded.

15.1.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will have to be in accordance with the Technical Requirements Manual; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

The activity released to the suppression pool chamber can be contained for some period of time. It is, therefore, assumed that the activity airborne above the suppression pool will be released under controlled conditions. The operator can choose to release the activity after decay has reduced the amount of activity to levels where the offsite dose consequence is minimal. For example, consider the case of a stuck open relief valve event with full MSIV closure which represents an upper bound on steam released to the suppression pool during an operational transient. The activity from this postulated event is released through the containment purge system at an assumed time of eight hours after the blowdown is complete (approximately 24 hours after the transient begins).

The containment airborne activity is discharged via the SGTS, with an assumed filter efficiency of 99 percent of the iodine activity. For this bounding example, the airborne activity above the suppression pool and the activity released to the environs are presented in Tables 15.1.-2 and 15.1-3 respectively. The offsite and control room radiological doses are given in Table 15.1.-4 for the maximum expected (realistic) and the design basis reactor coolant source terms. In both cases, the resultant doses are a small fraction of 10CFR20 limits detailed description of the control room model is provided in Appendix 15B. The input and assumptions used in the analysis are provided in Table 15.1-5.

15.1.5 SPECTRUM OF STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT IN A PWR

This event is not applicable to BWR plants.

15.1.6 INADVERTENT RHR SHUTDOWN COOLING OPERATION

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions no conceivable malfunction in the shutdown cooling system could cause temperature reduction.

If the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from maloperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from maloperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15E.1.6-1.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered while at power operation since the nuclear system pressure is too high to permit operation of the shutdown cooling mode of RHR.

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by maloperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

15.1.6.2.3 Effect of Single Failures and Operator Action

No single failures can cause this event to be more severe. If the operator takes action, the slow power rise will be controlled in the normal manner. If no operator action is taken, scram will terminate the power increase before thermal limits are reached and the operator will perform these actions in Table 15E.1.1-1. (See Appendix 15A for details.)

15.1.6.3 Core and System Performance

The increased subcooling caused by maloperation of the RHR shutdown cooling mode could result in a slow power increase due to the reactivity insertion. This power rise would be terminated by a high flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here.

15.1.6.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed, therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

Since this event does not result in any barrier failures, no analysis of radiological consequences is required for this event.

15.1.7 REFERENCES

15.1-1 Deleted

15.1-2 ANF-1358(P)(A) Revision 3, “The Loss of Feedwater Heating in Boiling Water Reactors,” Advanced Nuclear Fuels Corporation, September 2005.

15.1-3 PL-NF-89-005-A, “Qualification of Transient Analysis Methods for BWR Design and Analysis”, Pennsylvania Power & Light, Issue Date: July 1992

15.1-4 ANP-10300P-A Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome, January 2018.

15.1-5 XN-NF-80-19(P)(A) Volume 4 Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads” Exxon Nuclear Company, June 1986.

15.1-6 ANF-913(P)(A), Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” Advanced Nuclear Fuels Corporation, August 1990.

TABLE 15.1-1

SEQUENCE OF EVENTS FOR INADVERTENT
SAFETY RELIEF VALVE OPENING

TIME (SEC.)	EVENT
0	Opening of 1 safety relief valve, reaches full flow and remains open throughout the event.
1,200	Reactor scrammed on high suppression pool temperature. Technical Specification limit of 110°F. Closure of all MSIVs. Two loops of RHR suppression pool cooling placed into service.
5,200	Reactor depressurization initiated on high suppression pool temperature Technical Specification limit of 120°F.
59,200	Reactor depressurized to 14.7 psia, terminating blowdown through safety relief valve.

TABLE 15.1-2

SAFETY RELIEF VALVE OPENING EVENT
ACTIVITY ABOVE SUPPRESSION POOL (curies)

Isotope	Realistic	Design Basis
I-131	5.40E-01	2.22E+00
I-132	6.19E+00	2.54E+01
I-133	3.81E+00	1.54E+01
I-134	1.54E+01	6.19E+01
I-135	5.87E+00	2.38E+01
Kr-83m	2.53E+00	1.02E+01
Kr-85m	4.55E+00	1.83E+01
Kr-85	1.49E-02	6.01E-02
Kr-87	1.49E+01	6.01E+01
Kr-88	1.49E+01	6.01E+01
Kr-89	9.69E+01	3.91E+02
Xe-131m	1.12E-02	4.51E-02
Xe-133m	2.16E-01	8.72E-01
Xe-133	6.11E+00	2.46E+01
Xe-135m	1.94E+01	7.81E+01
Xe-135	1.64E+01	6.61E+01
Xe-137	1.12E+02	4.51E+02
Xe-138	6.63E+01	2.67E+02

TABLE 15.1-3

SAFETY RELIEF VALVE OPENING EVENT
ACTIVITY RELEASED TO THE ENVIRONS (curies)

Isotope	Realistic	Design Basis
I-131	5.24E-03	2.15E-02
I-132	5.55E-03	2.28E-02
I-133	2.91E-02	1.18E-01
I-134	2.75E-04	1.11E-03
I-135	2.53E-02	1.03E-01
Kr-83m	1.22E-01	4.92E-01
Kr-85m	1.32E+00	5.30E+00
Kr-85	1.49E-02	6.00E-02
Kr-87	1.90E-01	7.66E-01
Kr-88	2.11E+00	8.51E+00
Xe-131m	1.10E-02	4.42E-02
Xe-133m	1.94E-01	7.83E-01
Xe-133	5.84E+00	2.35E+01
Xe-135m	6.87E-09	2.76E-08
Xe-135m	8.90E+00	3.59E+01
Xe-138	4.20E-09	1.69E-08

TABLE 15.1-4

SAFETY RELIEF VALVE OPENING EVENT
RADIOLOGICAL DOSES (REM-TEDE)

Source Terms	EAB (REM-TEDE)	CRHE (REM-TEDE)
Realistic	1.87E-04	2.02E-04
Design Basis	7.54E-04	8.20E-04

Table 15.1-5

INADVERTENT SAFETY/RELIEF VALVE OPENING

Parameter	Design Basis Assumptions	Realistic Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source Term from Postulated Accident		
Core Thermal Power Level (MWt)	4032	4032
Fuel Damaged	None	None
Noble gas release rate (μCi/sec @ 30 min decay)	403,200	100,000
Iodine carryover fraction reactor water to steam (percent)	2% NWC 8% HWC	2% NWC 8% HWC
Radioiodine Chemical Species	95% Aerosol (CsI) 4.85% Elemental 0.15% Organic	95% Aerosol (CsI) 4.85% Elemental 0.15% Organic
II. Data and Assumptions Used to Estimate Activity Released		
Main Steam Mass Release (lbs)	3.5E+06	3.5E+06
Time for Release (seconds)	57,700	57,700
Turbine Design Flow Rate 9Lbm/hr)	1.69E+07	1.69E+07
Release Timing	Instantaneous release	Instantaneous release
SGTS Filter Bed Depth (inches)	8	8
SGTS Filter Bed Efficiency (percent)	99	99
III. Data and Assumptions Used to Evaluate Control Room Doses		
CRHE Volume (ft^3)	518,000	518,000
Control Room Free Air Volume (ft^3)	110,000	110,000
CRHE Isolation	No isolation assumed	No isolation assumed
CRHE Air Intake Flow	5229 to 6391 cfm	5229 to 6391 cfm
CRHE Ingress/Egress Flow	10 cfm	10 cfm
CRHE Unfiltered Inleakage	500 cfm	500 cfm
IV. Dispersion Data		
Site Boundary (meters)	549	549
X/Q's for Site Boundary	Table 2.3-92 (50 percentile)	Table 2.3-92 (50 percentile)
X/Q's for CRHE	Appendix 15B	Appendix 15B
V. Dose Data		
Method of Dose Conversion	Appendix 15B	Appendix 15B
Dose Conversion Factors	Appendix 15B	Appendix 15B
Doses	Table 15.1-4	Table 15.1-4