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DUKE POWER

October 15, 1991

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/91-09, Revision 1

Gentlemen:

Attached is Licensee Event Report 413/91-09, Revision 1 concerning POSSIBLE TECHNICAL SPECIFICATION VIOLATIONS RESULTING FROM IMPROPER OVERTEMPERATURE DELTA-TEMPERATURE CIRCUIT SCALING.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J.W. Hampton/RKC

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NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

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TITLE (4) TECHNICAL SPECIFICATION VIOLATIONS RESULTING FROM IMPROPER OVERTEMPERATURE DELTA TEMPERATURE CIRCUIT SCALING

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 7	1 7	9 1	9 1	0 0 9	0 1	1 0	1 5	9 1	CNS, Unit 2	0 5 0 0 0 4 1 4
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 0

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
C. L. Hartzell, Compliance Manager	8 0 3 8 3 1 - 3 6 6 5

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On July 17, 1991, with Units 1 and 2 in Mode 1, Power Operation, at 100 percent power, Maintenance Engineering Services (MES) personnel learned that the Overtemperature Delta-Temperature (OTDT) circuit would not work properly over the entire range. The gain associated with one of the OTDT calculation constants, K2, was applied to only one circuit card in each channel when the OTDT channels were calibrated at both McGuire Nuclear Station and Catawba. This would result in overranging the card, under certain conditions, for the current values of K2. This was discovered by MES personnel at McGuire during the week of July 8, 1991 during a scaling calculation for a new K2 value. It was not determined whether or not the OTDT channels were inoperable per Technical Specifications on July 17. It has been concluded that channels were inoperable in the past and that the plant would have been outside its design basis under some postulated conditions. This incident is attributed to defective vendor documentation and inappropriate action (due to inadequate procedure reviews). The gain value associated with the K2 constant was applied per vendor documentation, to the lead/lag circuit in the OTDT calibration procedures, resulting in potential saturation of the circuitry. Corrective actions included procedure and vendor documentation revisions, OTDT circuit modifications, and interim controls until hardware modifications were complete.

BACKGROUND

The Overtemperature Delta-Temperature (OTDT) function protects the core from Departure from Nucleate Boiling (DNB) for various transients involving certain combinations of pressure, power, coolant temperature, and axial power distribution provided that the transient is slow with respect to thermal delays associated with the Resistance Temperature Detectors [EIIS:XT] (RTDs) mounted in thermowells, and that pressure is between Pressurizer High and Low Pressure trips. The OTDT setpoint is continuously calculated for each Reactor Coolant [EIIS:AB] (NC) loop by solving an equation in Technical Specification (T/S) Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, Note 1. The OTDT function will trip the Reactor upon coincidence of two out of four channels reaching the trip setpoint. The OTDT setpoint is modified by average temperature, Pressurizer pressure, and delta flux (See Figure 1) by the application of penalties, and bonuses, depending on the magnitude and direction of the deviations of these parameters. T/S 3.3.1 requires a minimum of three OTDT channels to be operable.

The eight procedures used to initially set up the OTDT hardware in the 7300 Process Control Cabinets were IP/1,2/A/3222/76A, 76B, 76C, and 76D, Calibration Procedure for Delta-T/T-Ave Protection Channel I (II, III, IV). The gains for the OTDT circuit cards are specified in these procedures. The parameters required to be input into the OTDT hardware are specified in T/S Table 2.2-1, Note 1 and in the Precautions, Limitations and Setpoints for Nuclear Steam Supply Systems (PLS) Manual for Catawba Units 1 and 2. A scaling methodology was used to convert the required parameters to voltages, so that correct values would be obtained, module input or output limitations would not be exceeded, etc.

EVENT DESCRIPTION

On April 15, 1983, prior to initial operation of Catawba Nuclear Station, the value of one of the constants in the OTDT equation, K2, was changed from 1.327 to 2.401. This resulted in a change in the associated gain from 0.8847 volts per volt, to 1.6007 volts per volt. This change was made as a result of analyses for the decision to load an optimized fuel assembly core.

During the week of July 8, 1991, Maintenance Engineering Services (MES) personnel at McGuire Nuclear Station (MNS) were performing a scaling calculation to properly apply a new value for one of the constants in the T/S OTDT calculation, K2, to the OTDT hardware. This work was being performed in preparation for the upcoming Unit 1 Refueling Outage at MNS, in which a new type of fuel will be used. A Design Engineering safety analysis was performed in-house for the new type of fuel, which resulted in a new value for the K2 constant.

During the scaling process at MNS, MES personnel noted that application of the new K2 value would result in an earlier saturation or overranging of the operational amplifier (due to excessive gain) on the lead/lag circuit card of the OTDT circuits when the average NC temperature (Tave) exceeded a certain value. They asked Design Engineering whether or not the OTDT setpoint was to be applied over the full range of 530 to 630 degrees F Tave. On July 15, a Design Engineer went to McGuire to discuss the OTDT hardware setup. Design Engineering personnel stated that the setpoint was to be applied over the entire range. MES personnel at MNS then found that the current value of K2 would also result in excessive gain on the lead/lag circuit card above a Tave value of approximately 597 degrees F.

On July 16, Catawba MES personnel learned from McGuire MES personnel that it was possible that the K2 input to the OTDT calculation may not be properly applied over the full range. On July 17, at 0900 hours, with Catawba Units 1 and 2 in Mode 1. Power Operation, at 100 percent power, MES personnel at Catawba learned that the OTDT hardware would not function properly above a Tave of 592.5 degrees F. This would result in the OTDT circuit not applying an increasing temperature penalty to the setpoint for Tave above 592.5 degrees F. The range of the operating amplifier for all four OTDT channels is 0-10 volts, and with a gain setting of 1.6007V/V, the maximum output value required would be approximately 16 volts. The gain setting is directly correlated to the K2 value, which is 2.401%/degree F per the T/S OTDT equation. This condition was evaluated by Design Engineering and was discussed with the NRC on July 17. It was unknown on July 17 whether or not the OTDT channels were currently T/S inoperable. It was determined that hardware changes could be made to distribute the gain prior to the completion of a Design Engineering analysis.

On July 17, by 1845 hours, high priority work requests had been approved to redistribute the gain associated with the K2 constant from one to two cards in each of the eight (four per Unit) OTDT channels. Subsequently, on July 17, Technical Memorandums were issued which covered background information; required two operators per Unit to be in the Control Room; required that if rods [EIIS:ROD] move out uncontrolled, then to stop them and follow the Abnormal Procedures; required that if Reactor power exceeds 103%, then trip the Reactor; required that if power is reduced then hold at that power level until the OTDT hardware modifications are completed; specified not to run the Unit 1 Turbine [EIIS:TRB] Control [EIIS:IT] (ITE) System in Megawatt feedback; and required that control rods be in automatic, or, if in manual, that Tave remain below 592 degrees F. These measures were to apply until completion of the OTDT hardware modifications.

On July 17, by 2330 hours, the decision had been made to reduce Unit 1 power slightly because one channel of Overpower delta-Temperature (OPDT) had spiked (OPDT would be tripped as well as OTDT during the modifications to OTDT hardware). The purpose of this reduction would be to prevent receiving two OPDT tripped channels at the same time. By 0130 hours on July 18, Unit 1 power had been reduced to 98%. At 0130 hours, the modifications on OTDT

hardware began. By 0350 hours on July 19, the hardware modifications were complete, for both Units. By 0615 hours on July 19, Unit 1 had been returned to 100% power.

The modifications to the OTDT hardware involved distributing the gain associated with the K2 constant from one to two circuit cards in the 7300 Process Cabinets for each of the eight OTDT channels. This was done by replacing two resistors on the summing amp circuit card and adjusting a potentiometer on the lead/lag circuit card. The work was done under Exempt Change CE-3446 for Unit 1, and CE-3447 for Unit 2. Instrumentation and Electrical (IAE) procedures were revised to perform the work.

Over the next several days, DE personnel performed an assessment of past operating cycles. On July 23, 1991, at 0915 hours, it was concluded that it is possible that at some time in the past, the OTDT function was inoperable. At 1030 hours, NRC notification was made concerning OTDT past inoperability.

Design Engineering performed an assessment of operation with the degraded OTDT function and concluded on 8/21/91 that channels were inoperable in the past and that the plant would have been outside its design basis under some postulated conditions. These results were discussed at an Enforcement Conference with NRC on September 6, 1991.

CONCLUSION

This incident is attributed to a Design, Manufacturing, Construction/ Installation deficiency, due to erroneous vendor documentation of where to apply the gain associated with the K2 value in the PLS Manual. The PLS Manual, supplied by Westinghouse, lists where the parameters in the OTDT equation are to be applied. For the K2 value, only the lead/lag cards are listed: TY-412A, TY-422A, TY-432A, and TY-442A. Application of the gain in this manner will potentially saturate the lead/lag card circuitry. The Westinghouse PLS document was misleading as to the application of the K2 factor. It defined the K2 factor, and it listed a module where to apply the K2 factor. In reality, for any K2 factor exceeding 1.5 degrees per %, the gain has to be split between at least two modules. Westinghouse was aware of this potential problem as early as 1981. Other nuclear power plants received notification from Westinghouse, in the form of precautions within their purchased Scaling Manuals. Duke Power purchased scaling training from Westinghouse, and had a technical support contract with Westinghouse for I&C assistance after that time frame. However, Duke Power received no notification of the potential problem. In order to better understand the gain distribution, a Westinghouse supplied computer [EIIS:IMOD] scaling program was used (after discovery of the improper scaling).

This incident is attributed a contributing cause of Inappropriate Action, lack of attention to detail, due to inadequate reviews of the procedures used to set up the OTDT circuits. These reviews did not detect that the specified gain could possibly overrange the lead/lag circuit card. The gain specified

in procedures IP/1,2/A/3222/76A, 76B, 76C, and 76D was initially 1.6007 volts/volt for cards TY-412A, TY-422A, TY-432A, and TY-442A, respectively. This gain would result in an inaccurate temperature penalty input to the OTDT calculation for NC Tave temperatures above approximately 592.5 degrees F. The gain applied to these circuits was associated with the K2 value in the OTDT equation. Proper scaling would have ensured that the gain associated with the K2 value was not applied such that any of the components were overranged. These procedures were initially developed prior to Catawba startup. Values were taken from the PLS Manual, and procedures were reviewed by contract Westinghouse personnel.

The Catawba PLS Manual has been revised, adding "Interim-As-Built" sheets reflecting the gain distribution to include cards TY-411L, TY-421L, TY-431L, and TY-441L.

Corrective actions included evaluation of the condition by Design Engineering; revision of the OTDT calibration procedures; modification of OTDT hardware to distribute the gain associated with the K2 value; and revision of the Catawba PLS Manual to reflect the gain distribution. The need for and development of additional means to ensure proper scaling for future setpoint changes will be evaluated. All other Reactor protection and Engineered Safety Feature (ESF) circuits within the 7300 Process Control Cabinets at Catawba were checked to ensure that other circuits are not set up such that they could be overloaded under certain conditions. No similar situations were found. It was discovered during the investigation of this event that other Westinghouse nuclear power plants of like vintage also experienced similar scaling problems. They were made aware of the problem by notification from Duke Power via Nuclear Network entries, including the potential for circuit overranging resulting from gain distribution.

A review of the Operating Experience Program database showed that incidents involving deficient documentation and lack of attention to detail are recurring problems at Catawba. However, no other events have occurred during the past two years as a result of improper scaling.

CORRECTIVE ACTION

SUBSEQUENT

1) The OTDT hardware condition was evaluated by Design Engineering.

2) The OTDT calibration procedures were revised to properly set-up the OTDT circuits.

3) OTDT hardware was modified to distribute the gain associated with the K2 value.

4) The Catawba PLS Manual was revised to reflect the gain distribution for the K2 value.

5) Industry notification of the potential for circuit overranging resulting from gain distribution was made through Nuclear Network entries.

6) All Reactor protection and ESF circuits within the 7300 Process Control Cabinets were checked to ensure that other circuit cards are not set up such that they could be overloaded under certain conditions.

PLANNED

1) The development of additional means to further enhance proper scaling practices for future setpoint changes will be evaluated.

SAFETY ANALYSIS

The effect of the limited OTDT setpoint heatup penalty has been evaluated with respect to steady-state operation and FSAR Chapter 15 transients. The evaluation presented below was performed with the methodology used for the McGuire 1 Cycle 8 reload submittal and presented in Duke Power Company topical report DPC-NE-3002. The analytical results presented below were determined using physics parameter values which bound the current cycles of the McGuire and Catawba units as of mid-July, 1991. Although physics parameters vary from cycle to cycle, these cycles were considered sufficiently representative of previous fuel cycles so as to allow a meaningful evaluation.

Steady-state operation in the region in which the heatup penalty was limited was prevented by the DNB Parameters Technical Specification 3.2.5.a. For Catawba, this specification prohibits operation of the reactor with a control board meter indication of reactor vessel average temperature at greater than 592 degrees F, whereas the heatup penalty limit (saturation) would not have occurred until 592.5 degrees F.

The FSAR Chapter 15 transients which trip on OTDT in the DPC-NE-3002 methodology are:

Turbine trip (peak secondary pressure case)
Uncontrolled RCCA bank withdrawal at power (core cooling cases at slow withdrawal rates)
RCCA misoperation (single RCCA withdrawal)
CVCS malfunction that results in a decrease in boron concentration in the reactor coolant (power operation with manual rod control)
Inadvertent opening of a pressurizer safety or relief valve
Steam generator tube rupture

These transients were specifically evaluated or reanalyzed with the limited OTDT setpoint heatup penalty at the more conservative Catawba conditions, i.e., saturation occurring at 592.5 degrees F. For all the transients in which reanalyses were required, the limiting burnup is at the beginning of

core life. Therefore, Catawba 1 Cycle 6, which had a burnup of 22 FFPD on July 17, 1991, was used for the physics parameter input. The control rod drop time used in the reanalyses, 1.8 seconds to the top of the dashpot, is consistent with the slowest observed data from the testing required for Technical Specification 3.1.3.4. Except as noted for a particular reanalysis, other assumptions are consistent with the methodology presented in topical report DPC-NE-3002.

The peak secondary pressure case of the turbine trip transient is analyzed to demonstrate that the maximum pressure in the steam generator or the Main Steam System remains below 110% of the 1200 psia design pressure. If credit is taken for normal functioning of the steam line power-operated relief valves (PORVs), the maximum pressure is less than the limit even with a 3% drift (consistent with the proposed revision to Technical Specification Table 3.7-2) on the steam line safety valve setpoints. Based on this reanalysis, it is judged that an acceptable peak pressure would also be obtained without taking credit for the steam line PORVs if either 1) the current Technical Specification Table 3.7-2 steam line safety valve setpoint drift were used or if 2) credit were taken for reactor trip on turbine trip above the P-9 power level (69%). Therefore, it is judged that secondary pressure would have remained below the acceptance criterion for an actual turbine trip event.

The core cooling case of the uncontrolled RCCA bank withdrawal (UCBW) at power event is analyzed to demonstrate that the minimum departure from nucleate boiling ratio (DNBR) remains above the limit value, and therefore that fuel failures due to DNB do not occur. For this transient, a spectrum of initial power levels, burnups, and reactivity insertion rates are analyzed to ensure that the limiting accident is covered. The initial power levels considered are from 10% Rated Thermal Power (RTP) (the point at which the low setting power range high neutron flux trip is blocked) to 100% RTP. These same two extremes of initial power level are assumed in assessing the impact of the limited OTDT setpoint heatup penalty on the UCBW transient. Since the limited penalty reduces the effectiveness of this reactor trip function in terminating the heatup caused by this transient, it is assumed that any case which does not reach a reactor trip setpoint within twenty (20) minutes is ended by operator action.

The limiting case from 100% rated thermal power (RTP) is for a reactivity insertion rate just slow enough to avoid a power range high neutron flux (high setting) reactor trip. For faster insertion rates, the indicated neutron flux reaches the reactor trip setpoint before the actual core heat flux and temperature can reach values that are as limiting. For slower insertion rates, the reactor trip occurs, if at all, on high pressurizer level with an assumed response time of 2.0 seconds. This is due to the thermal expansion of the reactor coolant caused by the mismatch between the heat addition in the core and the heat removal in the steam generators. Slower insertion rates would cause less of a mismatch and thus take a longer time to reach the same limiting conditions. At the more negative temperature feedback conditions found at higher burnups, these limiting conditions are not reached prior to

the assumed operator action time. The limiting statepoint from this limiting rate was analyzed to determine a set of allowable core power distributions for which the minimum DNBR is at the limit value. These distributions were more restrictive than those analyzed for the McGuire 1 Cycle 8 reload safety analysis submittal. Therefore, the actual core power distribution for Catawba 1 Cycle 6 were generated at control rod positions corresponding to those which might be encountered during an UCBW transient. These actual distributions were compared to the allowable distributions to ensure that the power peaking in the actual core is low enough that the minimum DNBR would remain above the limit value. Therefore, even though the limit on the OTDT heatup penalty causes a more severe 100% RTP UCBW transient than the McGuire 1 Cycle 8 reload safety analysis submitted for inclusion into Chapter 15 of the FSAR, the acceptance criterion is still met for this case.

The UCBW transient dynamics for the case with a 10% RTP initial power level are similar to those discussed for the 100% RTP initial power level in the previous paragraph with two important exceptions. First, the Technical Specification allowed core power peaking is more severe at low power levels. Since the Duke Power Company transient analysis methodology conservatively assumes this relatively severe power peaking remains as the power level increases, a given thermal-hydraulic statepoint would result in a lower DNBR when the power distribution corresponding to low power operation is assumed. Second, the RCCA bank(s) which are assumed to be withdrawn from a 100% RTP initial power level are not the same ones which are assumed to be withdrawn from a 100% RTP initial power level. The RCCAs in the banks withdrawn from 10% RTP are not located near the power range neutron flux detectors. Therefore, as these banks are withdrawn, the ratio of the indicated (excore) flux to the actual (core) flux decreases. This makes the power range high neutron flux reactor trip less effective, allowing a more limiting thermal statepoint to be reached at reactor trip. These two effects cause a severe enough statepoint that the actual Catawba 1 Cycle 6 power distributions would result in a DNBR less than the limit value.

A DNBR less than the limit value gives the potential for fuel cladding failures due to DNB, which would release the activity in the fuel/clad gap to the coolant. However, the UCBW transient does not result in a rupture of the Reactor Coolant Pressure Boundary, does not result in the transport of reactor coolant inventory outside of the Containment via Emergency Core Cooling System operation in recirculation, and involves only limited release of reactor coolant to the Containment. Therefore, in the absence of steam generator tube leakage, the offsite dose due to this transient, even with large numbers of fuel failures, is insignificant. The actual operating history of McGuire and Catawba reveals only four occasions during which the tube leakage exceeded 10 gallons per day or 0.007 gallons per minute (gpm). At this level of primary-to-secondary leakage, even if all the fuel (100%) is assumed to fail due to DNB, the two hour exclusion area boundary (EAB) thyroid dose is 11 rem, which is within a small fraction (10%) of the 10 CFR 100 limits. At a more conservative value of tube leakage, e.g., 0.1 gpm, the two hour EAB thyroid dose for 10% fuel failures is 16 rem, which is also within a small fraction

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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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(10%) of the 10 CFR 100 limits. Therefore, even though the limit on the OTDT heatup penalty results in a UCBW from 10% RTP transient statepoint indicative of DNB, and therefore of potential fuel failures, the likely offsite dose associated with such an event would remain within 10% of 10 CFR 100.

The single RCCA withdrawal case of the RCCA misoperation transient is analyzed to demonstrate that 1) the potential fuel cladding failures due to DNB are limited to 5% of the fuel rods in the core and 2) that the offsite doses are within 10% of the 10 CFR 100 limits. The results of this transient are similar to the 100% RTP initial power level UCBW transient. Since the reactivity of a single RCCA is less than that of a RCCA bank, even the maximum reactivity insertion rate does not result in a reactor trip prior to the assumed 20 minute operator action time. Further, the negative temperature feedback present in the actual Catawba 1 Cycle 6 core, with respect to that assumed in the McGuire 1 Cycle 8 reload safety analysis, results in a lower core heat flux for a given temperature rise. At the limiting statepoint, the allowable core power distributions corresponding to a minimum DNBR at the limit value are therefore more easily met than those calculated for the McGuire 1 Cycle 8 reload safety analysis. The comparison of the actual McGuire 1 Cycle 8 power distributions to these more restrictive reload safety analysis allowable distributions demonstrated that fewer than 5% of the fuel rods would be in DNB for that core. Since the actual core power distributions are similar for McGuire 1 Cycle 8 and Catawba 1 Cycle 6, Catawba 1 Cycle 6 would be expected, when compared to the less restrictive allowable core power distributions calculated for this LER, to show even fewer fuel rod failures and to be easily with the 5% limit. Therefore, even though the limit on the OTDT heatup penalty causes a longer transient, which is terminated by operator action, the greater negative temperature feedback in an actual core causes a less severe transient than the corresponding McGuire 1 Cycle 8 reload safety analysis submitted for inclusion into Chapter 15 of the FSAR, and the acceptance criteria are still met for this case.

The CVCS malfunction that results in a decrease in the boron concentration in the reactor coolant (boron dilution) is analyzed to demonstrate that 1) DNB is avoided and 2) there is adequate time for the operator to terminate the dilution prior to the loss of shutdown margin. The case in which the dilution occurs at power operation with manual rod control has considerable margin to both acceptance criteria. If the limited OTDT setpoint heatup penalty is assumed to prevent an OTDT reactor trip, the power range high neutron flux (high setting) reactor trip will terminate power operation prior to DNB and with sufficient shutdown margin remaining to ensure adequate operator action time to terminate the dilution. Therefore the acceptance criteria for this event continue to be met even with the limited OTDT setpoint heatup penalty.

The inadvertent opening of a pressurizer safety valve and the steam generator tube rupture events are losses of RCS inventory events which cause an OTDT reactor trip because the depressurization penalty reduces the setpoint. The heatup penalty, which was affected by this event, plays essentially no role in causing the trip. Since the depressurization penalty functioned as designed

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during the time period in question, there was no impact on these two transients.

In summary:

1) Technical Specifications prevented steady-state operation at temperatures for which the OTDT trip heatup penalty was limited.

2) There was no impact on the turbine trip, single RCCA withdrawal, boron dilution, inadvertent opening of a pressurizer safety valve, and steam generator tube rupture events. The FSAR Chapter 15 acceptance criteria would have been met for these events.

3) The UCBW transient from the most likely initial condition (full power operation) would have met the FSAR Chapter 15 acceptance criterion.

4) The UCBW transient from a less likely low power initial condition could have resulted in DNB, with the potential for fuel failures and limited offsite doses.

The health and safety of the public were not affected by this incident.