TVA

Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

April 29, 2011

10 CFR 50.4(b)(6)
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 - Response to Requests for Additional Information (RAIs) Regarding Inadvertent ECCS Actuation Analysis, and Chemical & Volume Control System Malfunction Analysis

This letter provides responses to RAIs regarding: (1) Inadvertent ECCS Actuation Analysis and (2) Chemical & Volume Control System Malfunction Analysis. These RAIs were received during the recent NRC Audit at the Westinghouse Electric Company LLC offices in Rockville, Maryland, held during the week of March 14, 2011.

Enclosure 1 provides the response to the Inadvertent ECCS Actuation analysis portion of the RAIs.

Enclosure 2 provides the response to the Chemical & Volume Control System Malfunction Analysis portion of the RAIs.

Enclosure 3 provides a list of the new regulatory commitments contained in this letter. If you have any questions, please contact Bill Crouch at (423) 365-2004.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 29th day of April, 2011.

Respectfully,

David Stinson
Watts Bar Unit 2 Vice President

A002
NRR

<u>Enclosures</u>:

1. Response to RAI Concerning Watts Bar Unit 2 Inadvertent ECCS Actuation Analysis

2. Response to RAI Concerning Watts Bar Unit 2 Chemical & Volume Control System (CVCS) Malfunction Analysis

3. List of Commitments

<u>Attachments</u>:

1. Proposed Markup to Unit 2 FSAR - Incorporates Analysis for Inadvertent ECCS Actuation

2. Proposed Markup to Unit 2 FSAR - Incorporates Analysis for Inadvertent CVCS Malfunction

cc (Enclosures):

U. S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

NRC Resident Inspector Unit 2
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

ENCLOSURE 1

RESPONSE TO RAI CONCERNING WATTS BAR UNIT 2
INADVERTENT ECCS ACTUATION ANALYSIS

During the audit, TVA was asked to reanalyze the Inadvertent ECCS event to be consistent with the concerns documented in NRC Regulatory Issue Summary 2005-29.

To address this request, Westinghouse performed a review of the analysis presented in the licensing application and identified sources of margin. The original analysis modeled pressurizer backup heater actuation on high level deviation, which results in additional coolant heatup and expansion in the pressurizer. This control system function has been defeated for both Watts Bar units. The reanalysis reflects this change. Second, the original analysis modeled a bounding initial pressurizer level uncertainty. For the reanalysis, this uncertainty was reduced to the actual plant calculated pressurizer level uncertainty. Finally, the original analysis modeled a bounding ECCS flow rate. The reanalysis used a Unit 2 specific maximum ECCS flow rate, which is conservative but lower than what was used in the original analysis.

The results of the updated analysis demonstrate that the pressurizer reaches a peak water volume of 1,732 ft^3 at approximately 667 seconds, compared to the total pressurizer volume of 1,846 ft^3 (including the surge line volume). The pressurizer does not go water solid during the event, assuming operator action time at 10 minutes to terminate injection flow. Therefore, neither the pressurizer power-operated relief valves (PORVs) nor the pressurizer safeties open while a water-solid condition exists in the pressurizer and the non-escalation acceptance criterion is met.

This analysis will be incorporated into the Unit 2 FSAR as shown in Attachment 1.

ENCLOSURE 2

RESPONSE TO RAI CONCERNING WATTS BAR UNIT 2 CHEMICAL & VOLUME CONTROL SYSTEM (CVCS) MALFUNCTION ANALYSIS

During the audit, TVA was asked to perform a CVCS malfunction analysis as required by NUREG-800, SRP 15.5.1-15.5.2.

Westinghouse performed the requested analysis. The analysis documents two CVCS Malfunction cases:

1. maximum flow from one charging pump with letdown isolated, and
2. maximum flow from two charging pumps with 75 gpm of letdown.

The analysis assumes:

- No reactor trip.
- The flow source is assumed to be at the RCS boron concentration.
- The same pumps are providing the flow as in the Inadvertent ECCS Event, but the flow path has a higher resistance than the Safety Injection flow path. Thus, the CVCS Malfunction flow rates are lower than the Inadvertent ECCS flow rates.
- Alarm actuation alerts the operator 60 seconds after event initiation.
- Operator terminates charging flow 10 minutes after an alarm notifies the operator.

Results:

The pressurizer level increases as a result of the injected flow. In the case with one charging pump operating, the pressurizer reaches a peak water volume of 1,664 ft^3, and in the case with two charging pumps operating, the peak pressurizer water volume is 1,635 ft^3. Since the pressurizer does not fill in either case, there can be no water relief through either the PORVs or the Pressurizer Safety Relief Valves. The details of the assumptions and analysis results are provided in Attachment 2.

A future amendment to the Unit 2 FSAR will incorporate this analysis into the FSAR as shown in Attachment 2.

ENCLOSURE 3

LIST OF COMMITMENTS

1. A future amendment to the Unit 2 FSAR will incorporate the Inadvertent ECCS Actuation analysis.

2. A future amendment to the Unit 2 FSAR will incorporate the CVCS Malfunction analysis.

ATTACHMENT 1

Proposed Markup to Unit 2 FSAR

Incorporates Analysis for Inadvertent ECCS Actuation

This mark-up is applicable to Watts Bar Unit 2 only.

Table 15.1-2 Summary Of Initial Conditions And Computer Codes Used (Continued)
(Page 2 of 4)

FAULTS	COMPUTER CODES UTILIZED	REACTIVITY COEFFICIENTS ASSUMED FOR: MODERATOR TEMPERATURE (Δk/°F)	MODERATOR DENSITY (Δk/gm/cc)	DOPPLER	INITIAL NSSS THERMAL POWER OUTPUT ASSUMED[1] (MWt)
CONDITION II (Cont'd)					
Accidental Depressurization of the Main Steam System	Accident evaluated; bounded by major rupture of a steam pipe				
Accidental Depressurization of the Reactor Coolant System	LOFTRAN		0.00	upper[2]	3425
Inadvertent Operation of ECCS During Power Operation	LOFTRAN	---	0.00 and 0.43	lower and upper[2]	~~3425~~ 3475[5]
CONDITION III					
Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling	NOTRUMP, LOCTA-IV				3411[5]
Inadvertent Loading of a Fuel Assembly into an Improper Position	LEOPARD, TURTLE	---	Minimum	NA	3425
Complete Loss of Forced Reactor Coolant Flow	VIPRE-01, FACTRAN, LOFTRAN	---	0.00	upper[2]	3475
Waste Gas Decay Tank	NA	---	NA	NA	3579
Single RCC Assembly Withdrawal at Full Power	TURTLE, THINC, LEOPARD	---	NA	NA	3425

Out of Scope

Out of Scope

Results OUT OF SCOPE

Since the conditions above for an accidental depressurization of the main steam system are significantly less limiting than those for the main steam line rupture (MSLB, 15.4.2) transient from HZP conditions and since these events are analyzed utilizing similar methodology, the analysis for the MSLB transient is used to bound the accidental depressurization of the main steam system event. This approach is supported by the fact that the maximum return to power for steam release transient is much lower than that for the HZP MSLB event. Hence, minimum DNBR is not a concern under these conditions.

15.2.13.3 Conclusions

The analysis shows that the criteria stated earlier in this section are satisfied since a DNBR less than the limiting value does not exist.

15.2.14 Inadvertent Operation of Emergency Core Cooling System

This analysis was performed after the boron injection tank (BIT) and associated 900 gallons of 20,000 ppm boron were deleted from the Watts Bar design basis. Therefore, the BIT is not referred to in this section.

15.2.14.1 Identification of Causes and Accident Description

Spurious Emergency Core Cooling System (ECCS) operation at power could be caused by operator error or a false electrical actuating signal. Spurious actuation may be assumed to be caused by any of the following:

(1) High containment pressure

(2) Low pressurizer pressure (above Permissive P11)

(3) Low steamline pressure (above Permissive P11)

(4) Manual actuation

Following the actuation signal, the suction of the centrifugal charging pumps is diverted from the volume control tank to the refueling water storage tank.

The charging pumps then force concentrated (3300 ppm) boric acid solution from the RWST, through the common injection header and injection lines and into the cold leg of each reactor coolant loop. The safety injection pumps also start automatically, but provide no flow when the reactor coolant system is at normal pressure. The passive injection system and the low head system provide no flow at normal reactor coolant system pressure.

A safety injection signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates a safety injection signal will also produce a reactor trip. Therefore, two different courses of events are considered.

~~*Maximum RWST boric acid solution is conservative for this event analysis. A value of 2700 ppm is modeled in the analysis, however, evaluations have been performed to support a maximum concentration of 3300 ppm.~~

(1) Case A - Trip occurs at the same time spurious injection starts.

The operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the safety injection signal should be blocked. For a spurious occurrence, the operator would stop the safety injection and maintain the plant in the hot shutdown condition. If the ECCS actuation instrumentation must be repaired, future plant operation will be in accordance with the Technical Specifications.

(2) Case B -The reactor protection system produces a trip later in the transient.

The reactor protection system does not produce an immediate trip, and the reactor experiences a negative reactivity excursion due to the injected boron causing a decrease in reactor power. The power mismatch causes a drop in primary coolant temperature and coolant shrinkage. Pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load when the turbine throttle valve is fully open. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low pressure trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history which affects initial boron concentration, rate of change of boron concentration, Doppler and moderator coefficients.

Recovery from this incident for Case B is made in the same manner described for Case A. The only difference is the lower T_{avg} and pressure associated with the power mismatch during the transient. The time at which reactor trip occurs is of no concern for this occurrence. At lower loads coolant contraction will be slower resulting in a longer time to trip.

15.2.14.2 Analysis of Effects and Consequences

Method of Analysis

The spurious operation of the safety injection system is analyzed by employing the detailed digital computer program LOFTRAN[5]. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the safety injection system. The program computes pertinent plant variables including temperatures, pressures, and power level.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

(a) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

(b) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and

(c) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

To address criterion (c), ~~Westinghouse currently uses~~ the more restrictive criterion that ~~a water-solid pressurizer condition be precluded when the pressurizer is at or above the set pressure of the pressurizer safety relief valves (PSRVs).~~ This addresses any concerns regarding subcooled water relief through the ~~plant PSRVs~~ which are not qualified for this condition. ~~Should water relief through the pressurizer power-operated relief valves (PORVs) occur, the PORV block valves would be available, following the transient, to isolate the RCS.~~

pressurizer power-operated relief valves (PORVs) do not open while a water-solid condition exists in the pressurizer is used.

pressurizer PORVs or pressurizer safety relief valves (PSRVs) and the downstream piping,

The inadvertent ECCS actuation at power event is analyzed to determine both the minimum DNBR value and maximum pressurizer water volume. The most limiting case with respect to DNB is a minimum reactivity feedback condition with the plant assumed to be in manual rod control. Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits.

For maximizing the potential for pressurizer filling, the most limiting case is a maximum reactivity feedback condition with an immediate reactor trip, and subsequent turbine trip, on the initiating SI signal. The transient results are presented for each case.

Assumptions

(1) Initial Operating Conditions

The DNB case is analyzed with the Revised Thermal Design Procedure as described in WCAP-11397-P-A[18]. Initial reactor power, RCS pressure, and temperature are assumed to be at the nominal full power values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference [18]. For the pressurizer filling case, initial conditions with uncertainties in their worst possible direction on power, vessel average temperature, pressurizer pressure, and pressurizer level are assumed in order to maximize the rate of coolant expansion and minimize the size of the steam bubble.

(2) Moderator and Doppler Coefficients of Reactivity

The minimum DNBR case is evaluated at beginning of life (BOL) conditions, so a low BOL moderator temperature coefficient and a low absolute value

Doppler power coefficient are assumed. For the pressurizer ~~pressure~~ filling case, conservative maximum feedback coefficients consistent with end of life operation are assumed.

(3) Reactor Control

For the minimum DNBR case (without direct reactor trip on SI), the reactor is assumed to be in manual rod control. For the pressurizer filling case, a reactor trip is assumed to occur coincident with initiation of the transient.

(4) Pressurizer Pressure Control

Pressurizer heaters are assumed to be inoperable for the minimum DNBR case, since this yields a higher rate of pressure decrease. The opposite is assumed for the pressurizer filling case, in which the operation of the pressurizer heaters has been found to result in an increase in the pressurizer filling rate.

PORVs are ~~assumed as an automatic pressure control function~~ operable for both the minimum DNBR and pressurizer filling cases. For the minimum DNBR case, maintaining a low pressurizer pressure is conservative. For the pressurizer filling case, availability of the PORVs may provide~~s~~ earlier steam relief and therefore maximize~~s~~ the pressurizer insurge. However, ~~since the pressurizer filled in the WBN analysis, the final pressurizer case assumed that the PORVs are unavailable. This maximizes the pressure, which is conservative for the purpose of determining whether or not the safety valves actuate~~ the PORV opening setpoint is not reached in this analysis.

Pressurizer spray is assumed available to minimize pressure for the minimum DNBR case and to increase the rate of the pressurizer level increase for the pressurizer filling case.

(5) Boron Injection

At the initiation of the event, two centrifugal charging pumps inject borated water into the cold leg of each loop. In addition, flow is included to account for the potential operation of the positive displacement charging pump (PDP) for the DNBR case. However, this analysis remains valid although the PDP has been abandoned and is no longer used for normal operation. No PDP flow is assumed for the overfill case since the pump is not used for normal operation.

(6) Turbine Load

For the minimum DNBR case (without direct reactor trip/turbine trip on SI), the turbine load remains constant until the governor drives the throttle valve wide open. After the throttle valve is fully open, turbine load decreases as steam pressure drops. In the case of pressurizer filling, the reactor and turbine both trip at the time of SI actuation with the turbine load dropping to zero simultaneously.

(7) Reactor Trip

Reactor trip is initiated by low pressure at 1925 psia for the minimum DNBR case. The pressurizer filling case assumes an immediate reactor trip on the initiating SI signal.

(8) Decay Heat

The decay heat has no impact on the DNB case (i.e., minimum DNBR occurs prior to reactor trip). For the pressurizer filling case, the availability of decay heat and its expansion effects on the RCS liquid volume is considered. Core residual heat generation is based on the 1979 version of ANSI 5.1[14] assuming long-term operation at the initial power level preceding the trip is assumed.

(9) Operator Actions

Operator action to terminate safety injection flow is assumed 10 minutes from event initiation, and thereby, mitigates the event.

(10) Auxiliary Feedwater System

For the pressurizer filling case only, the AFW System is assumed to actuate on the initiating SI signal. The AFW flow provides additional RCS cooling which slows the pressurizer in~~x~~surge.

Results

The transient responses for the ~~minimum DNBR and~~ pressurizer filling case~~s~~ are shown in Figures 15.2-42a through 15.2-42c. Table 15.2-1 shows the calculated sequence of events. for both the minimum DNBR case and the pressurizer filling case.

Minimum DNBR Case:

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The reactor trips on low pressurizer pressure. After trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat. The DNBR remains above its initial value throughout the transient.

Pressurizer Filling Case:

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the RCS. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the RCS heatup, due to residual RCS heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around.

SI flow termination at 10 minutes prevents the pressurizer from filling.

Pressurizer water level then increases throughout the transient. Spray flow helps to condense the pressurizer steam bubble, causing a pressurizer insurge and minimizing pressurizer pressure.The ECCS injection flow is terminated via operator action in accordance with plant emergency procedures and the increase in pressurizer level stops. ~~Although the pressurizer becomes water solid just prior to SI termination, the maximum pressure reached is below the pressurizer safety valve opening setpoint.~~ As such, the integrity of the ~~safety valves is~~ not compromised.

PORVs and the PSRVs are

Following the analyzed portion of the transient, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain generator level through control of the main or auxiliary feedwater system. Any action required of the operator to maintain the plant in a stabilized condition is in a time frame in excess of ten minutes following reactor trip.

15.2.14.3 Conclusions

Results of the analysis show that spurious ECCS operation without immediate reactor trip does not present any hazard to the integrity of the RCS with respect to DNBR. The minimum DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the RCS. If the reactor does not trip immediately, the low pressurizer pressure reactor trip will provide protection. This trips the turbine and prevents excess cooldown, which expedites recovery from the incident.

With respect to pressurizer filling, ~~although the pressurizer becomes water solid just prior to SI termination, the maximum pressure reached is below the opening pressure of the safety valves.~~ As such, the ~~safety valves~~ do not pass water and their integrity is not compromised. Termination of ECCS injection via operator action in accordance with plant emergency procedures~~,~~ stops the further increase in pressure, thus preventing the safety valves from opening. This precludes possible damage to the valves which could potentially generate a more serious plant condition.

SI flow is terminated in sufficient time to prevent the pressurizer from going water solid.

PORVs and the PSRVs

References

OUT OF SCOPE

(1) Gangloff, W. C., "An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors," WCAP-7486-L (Proprietary), March 1971 and WCAP-7486 (Non-Proprietary), May 1971 .

(2) "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August 1974.

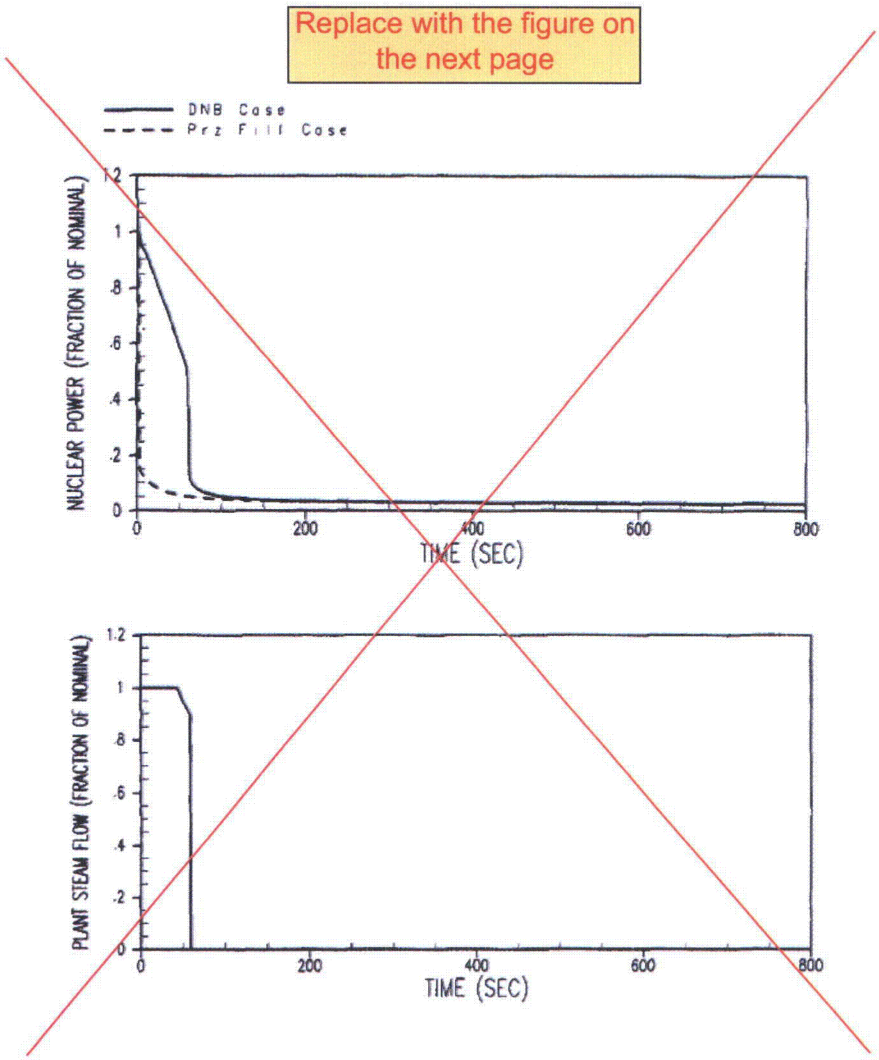
(3) Risher, D. H. Jr. and Barry, R. F., "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," WCAP-7979-PA (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

(4) Hargrove, H. G., "FACTRAN, A FORTRAN IV Code for Thermal Transients in a $U0_2$ Fuel Rod," WCAP-7908-A, December 1989.

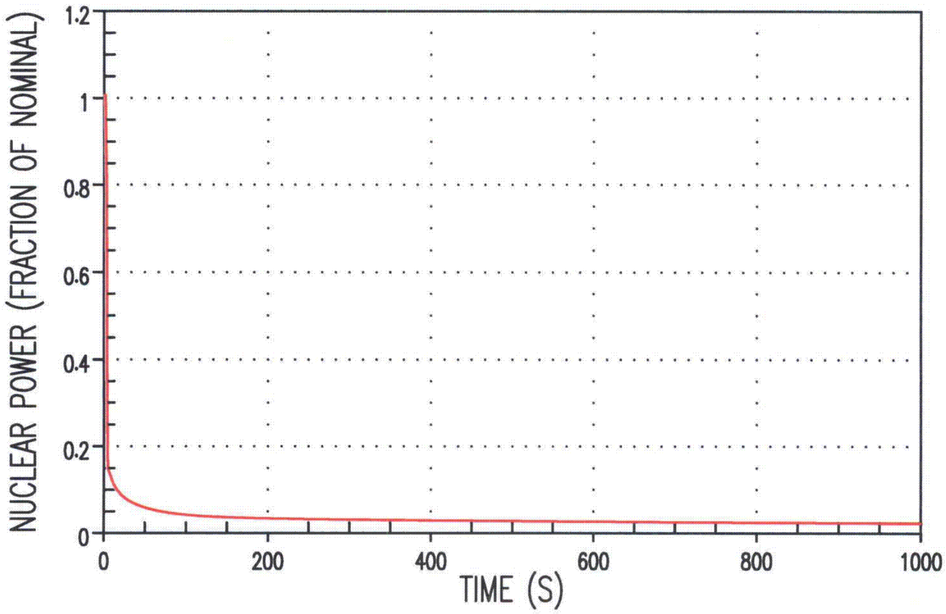
Out of Scope

Table 15.2-1 Time Sequence Of Events For Condition II Events (Page 5 of 5)

Accident	Event	Time (sec.)
Accidental Depressurization of the Reactor Coolant System	Inadvertent opening of one pressurizer safety valve	0.0
	OTΔT reactor trip setpoint reached	32.3
	Rods begin to drop	33.8
	Minimum DNBR occurs	34.4
Inadvertent Operation of ECCS During Power Operation DNBR Case:	Charging pumps begin injecting borated water; neutron flux starts decreasing	0.0
	Steam flow starts decreasing	44
	Low pressurizer pressure reactor trip setpoint reached	56
	Rods begin to drop	58
	Minimum DNBR occurs	(1)
Pressurizer Filling Case:	Charging pumps begin injecting borated water; reactor trip on 'S' signal; rod motion begins	0.0
	~~Pressurizer Fills~~	~~575~~
	Operator terminates injection flow	600
	~~MaximumRCS pressure occurs~~	~~602~~
(1)DNBR does not decrease below its initial value.	Maximum pressurizer water level occurs	667



~~**Figure 15.2-42a Inadvertent Operation of Emergency Core Cooling System - Nuclear Power& Steam Flow Response**~~



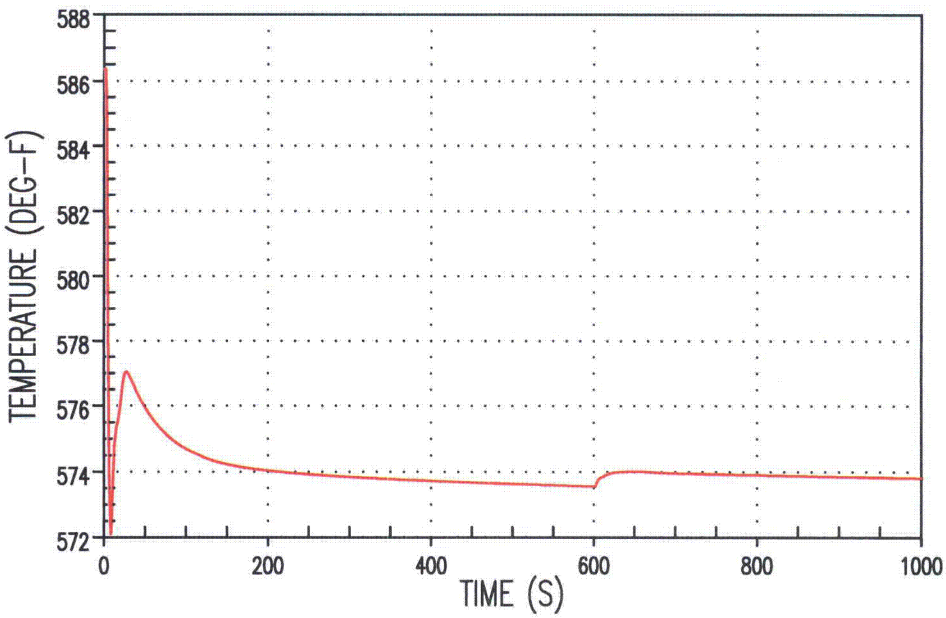


Figure 15.2-42a Inadvertent Operation of the Emergency Core Cooling System - Nuclear Power and Core Average Temperature Response

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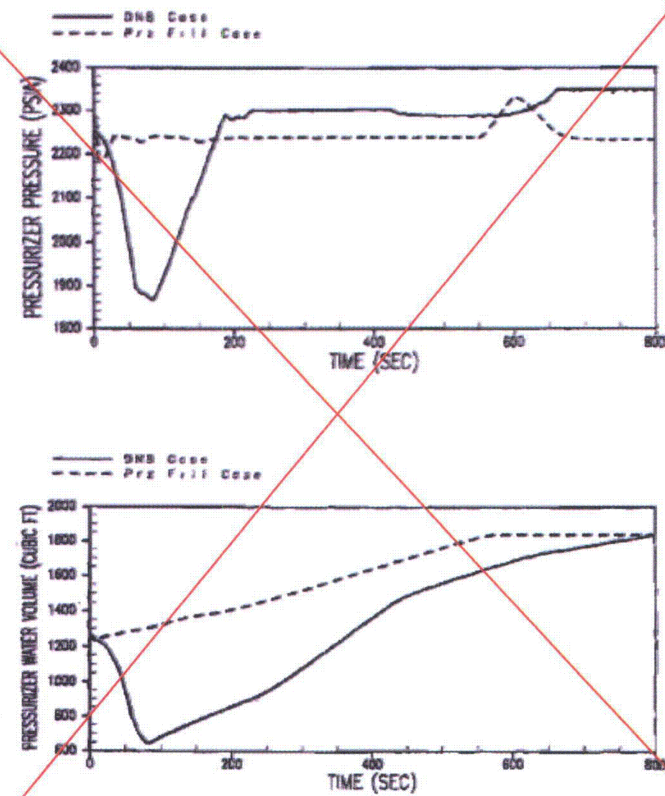


~~Figure 15.2-42b Inadvertent Operation of Emergency Core Cooling System - Pressurizer Pressure & Water Volume Response~~

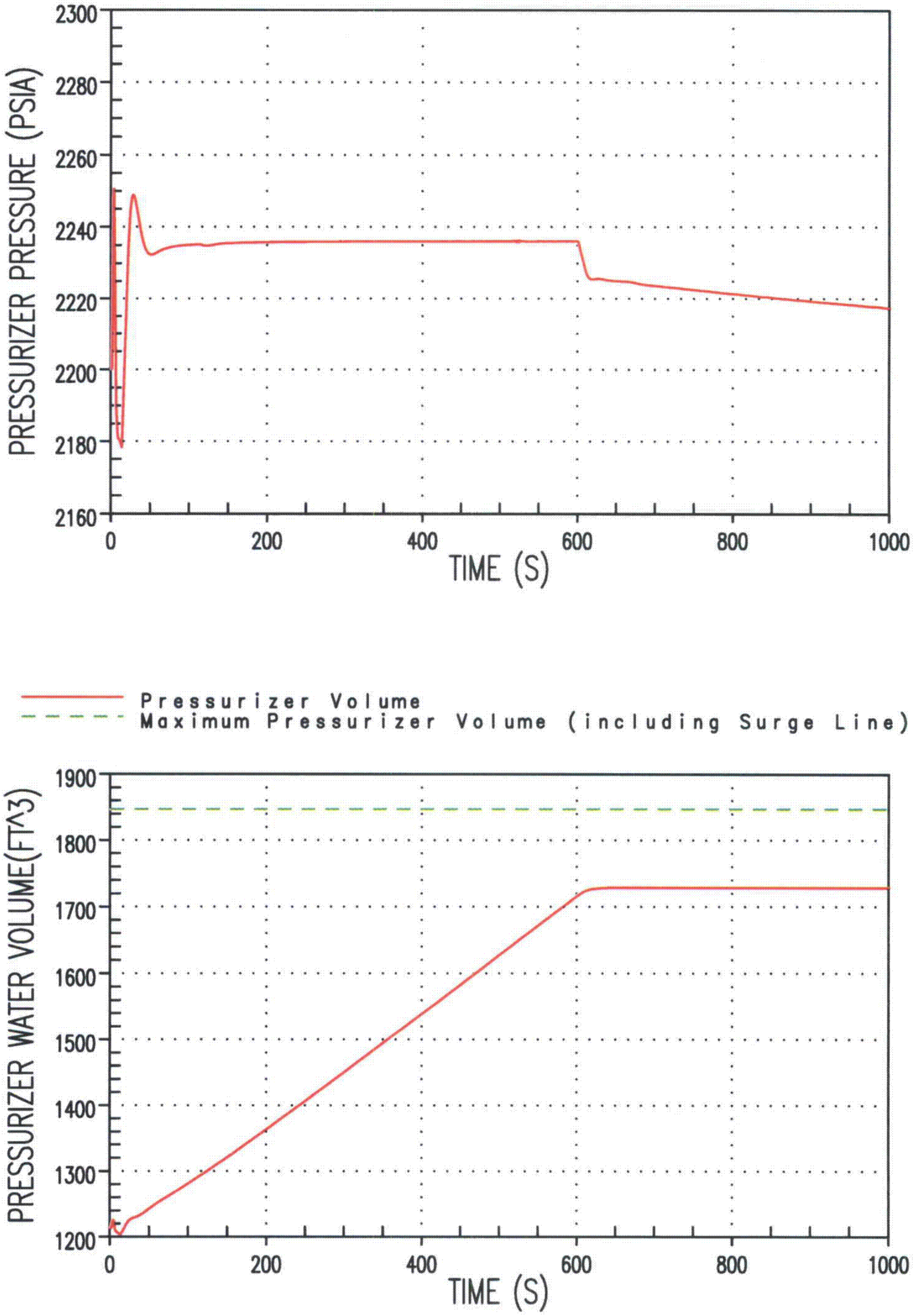
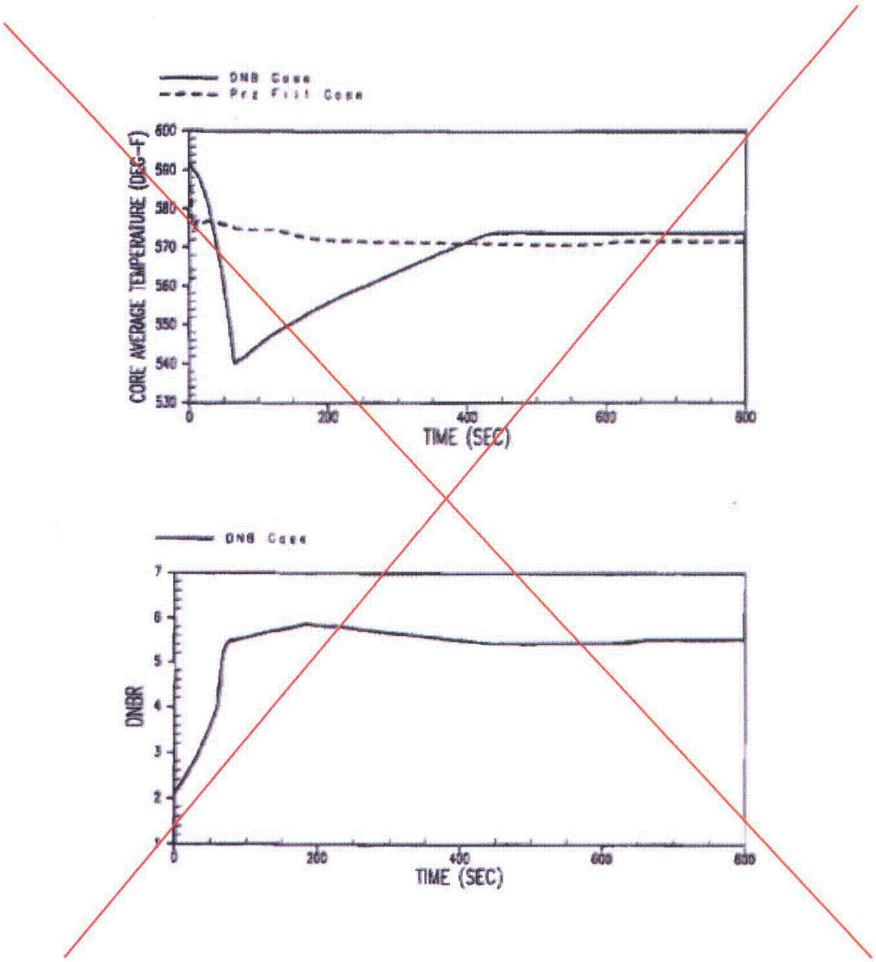


Figure 15.2-42b Inadvertent Operation of the Emergency Core Cooling System – Pressurizer Pressure and Pressurizer Water Volume Response

Replace with the figure on the next page



~~**Figure 15.2-42c Inadvertent Operation of Emergency Core Cooling System - Core Average Temperature And DNBR Response**~~

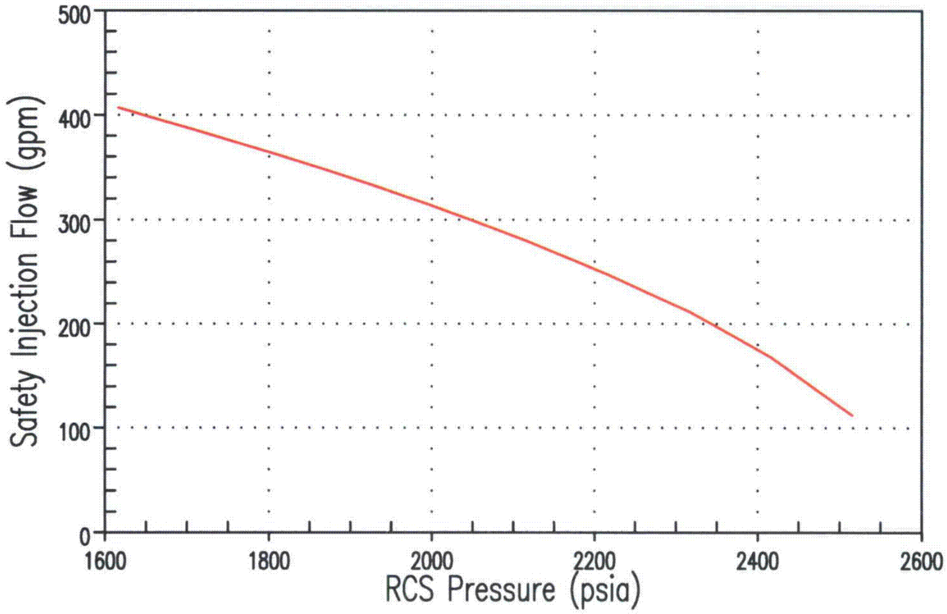


Figure 15.2-42c Inadvertent Operation of the Emergency Core Cooling System – Maximum Emergency Core Cooling System Flow Rate

ATTACHMENT 2

Proposed Markup to Unit 2 FSAR

Incorporates Analysis for Inadvertent CVCS Malfunction

15.2.15. Chemical and Volume Control System Malfunction During Power Operation

15.2.15.1 Identification of Causes and Accident Description

Increases in reactor coolant inventory caused by a malfunction of the chemical and volume control system may be postulated to result from operator error or a control signal malfunction. Transients examined in this section are characterized by increasing pressurizer level, increasing pressurizer pressure, and a constant boron concentration. The transients analyzed in this section are done to demonstrate that there is adequate time for the operator to take corrective action to ensure that the integrity of the pressurizer Power Operated Relief Valves (PORVs) and the Pressurizer Safety Relief Valves (PSRVs) is maintained (i.e., the valves do not actuate with the pressurizer in a water-solid condition). An increase in reactor coolant inventory, which results from the addition of cold, unborated water to the RCS, is analyzed in Section 15.2.4, Uncontrolled Boron Dilution.

The most limiting CVCS Malfunction case would result if charging was in automatic control and the pressurizer level channel being used for charging control failed in a low direction. This would cause maximum charging flow to be delivered to the RCS and letdown flow to be isolated. The worst single failure for this event would be a second pressurizer level channel failing in an as-is condition or a low condition. This will defeat the reactor trip on two-out-of-three high pressurizer level channels. To ensure that the integrity of the PORVs and the PSRVs is maintained, the operator must be relied upon to terminate charging.

During a CVCS Malfunction event, several main control board alarms could be generated to alert the operator, including the following:

- High charging flow alarm
- High pressurizer water level alarm
- Pressurizer water level deviation alarm
- Low VCT level alarm

15.2.15.2 Analysis of Effects and Consequences

Method of Analysis

The CVCS malfunction is analyzed using the LOFTRAN computer code (WCAP-7907-P-A). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

A Chemical and Volume Control System Malfunction at power event is classified as a Condition II event, a fault of moderate frequency. The criteria established for Condition II events include the following:

- Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,
- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

Of these, the limiting criterion is that the event will not propagate to a more serious event. To address this criterion, Westinghouse currently uses the more restrictive criterion that the pressurizer Power Operated Relief Valves (PORVs) do not open while a water-solid condition exists in the pressurizer. This addresses any concerns regarding subcooled water relief through the pressurizer PORVs or Pressurizer Safety Relief Valves (PSRVs) and the downstream piping which may not be qualified for this condition.

The analysis assumptions are the same as those discussed in Section 15.2.14.2 for the pressurizer filling case with a few exceptions:

- No reactor trip is assumed.
- The flow source is assumed to be at the RCS boron concentration.
- The same pumps are providing the flow as in the Section 15.2.14 event, but the flow path has a higher resistance than the Safety Injection flow path. Thus, the CVCS Malfunction flow rates are lower than the Inadvertent ECCS flow rates.
- Alarm actuation alerts the operator 60 seconds after event initiation.
- Operator terminates charging flow 10 minutes after an alarm notifies the operator.

Cases are examined with flow from both one and two centrifugal charging pumps to determine the time available for the operators to take the necessary corrective actions to maintain the integrity of the PORVs and PSRVs. The scenario analyzed with two charging pumps operating is slightly different than the one charging pump scenario. In the two-pump scenario, it takes two failures to have two charging pumps operating at maximum capacity. Letdown isolation would require a third failure, so letdown is not isolated in the two-pump case. Minimum letdown flow is 75 gpm so the net inventory addition is decreased by 75 gpm for the two-pump case.

Results

The transient responses for the limiting CVCS system malfunction cases are shown in Figures 15.2.15-1 through Figure 15.2.15-4. Table 15.2.15-1 shows the calculated sequence of events. In all the cases analyzed, core power and RCS temperatures remain relatively constant.

The pressurizer level increases as a result of the injected flow. In the case with one charging pump operating, the pressurizer reaches a peak water volume of 1664 ft^3, and in the case with 2 charging pumps operating, the peak pressurizer water volume is 1635 ft^3. Since the pressurizer does not fill in either case, there can be no water relief through either the PORVs or the PSRVs.

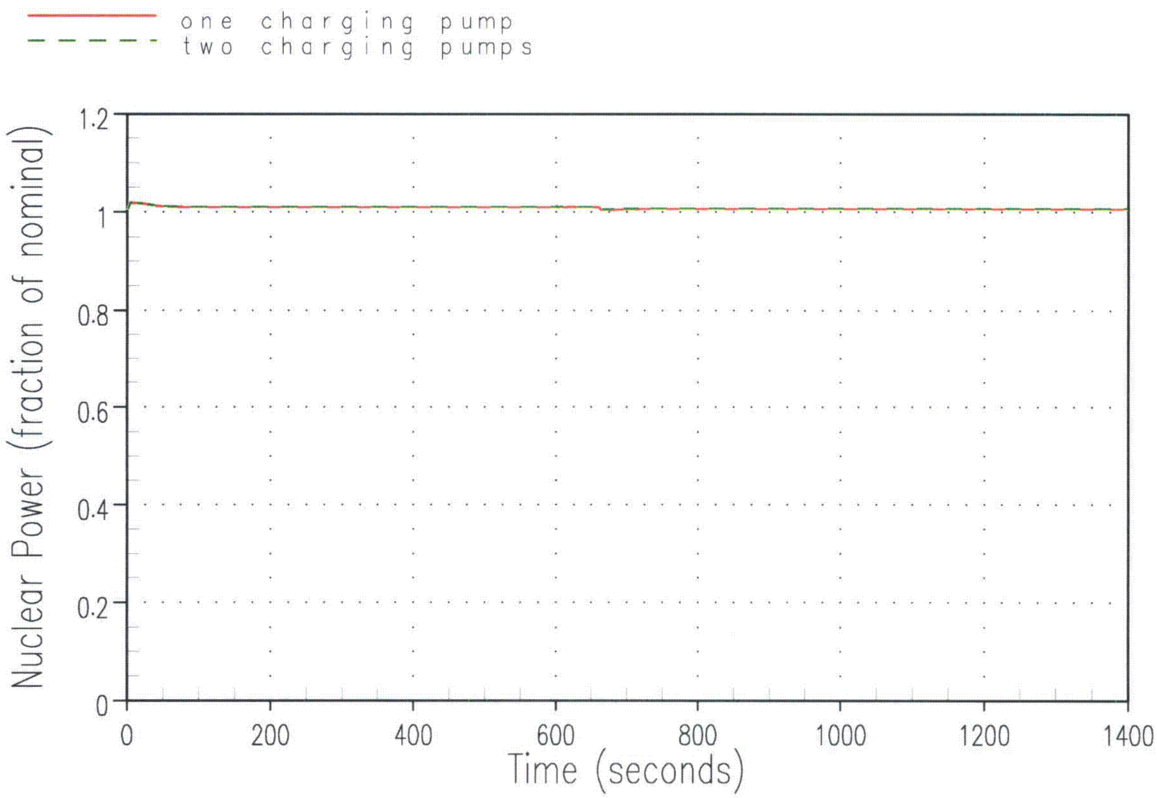
Figures 15.2.15-1 through 15.2.15-4 provide transient information for both the one-pump and two-pump cases and Table 15.2.15-1 shows a sequence of events.

15.2.15.3 Conclusions

With respect to not creating a more serious plant condition, water relief out of the PORVs and PSRVs will not occur during a CVCS Malfunction event because operator action to terminate the charging flow occurs early enough to prevent a water-solid pressurizer. The sequence of events presented in Table 15.2.15-1 shows the operators have sufficient time to take corrective action.

TABLE 15.2.15-1 TIME SEQUENCE OF EVENTS FOR CVCS MALFUNCTION

Accident	Event	Time (sec)
CVCS malfunction, one pump operating	Maximum charging flow initiated / letdown isolated	0.0
	An annunciator on the control board alerts the operator that an event is occurring	60.0
	Operator terminates charging flow.	660.0
	Peak pressurizer water volume is reached.	1479.1
CVCS malfunction, two pumps operating	Maximum charging flow initiated from two charging pumps	0.0
	An annunciator on the control board alerts the operator that an event is occurring	60.0
	Operator terminates charging flow	660.0
	Peak pressurizer water volume is reached	688.2



WATTS BAR NUCLEAR PLANT

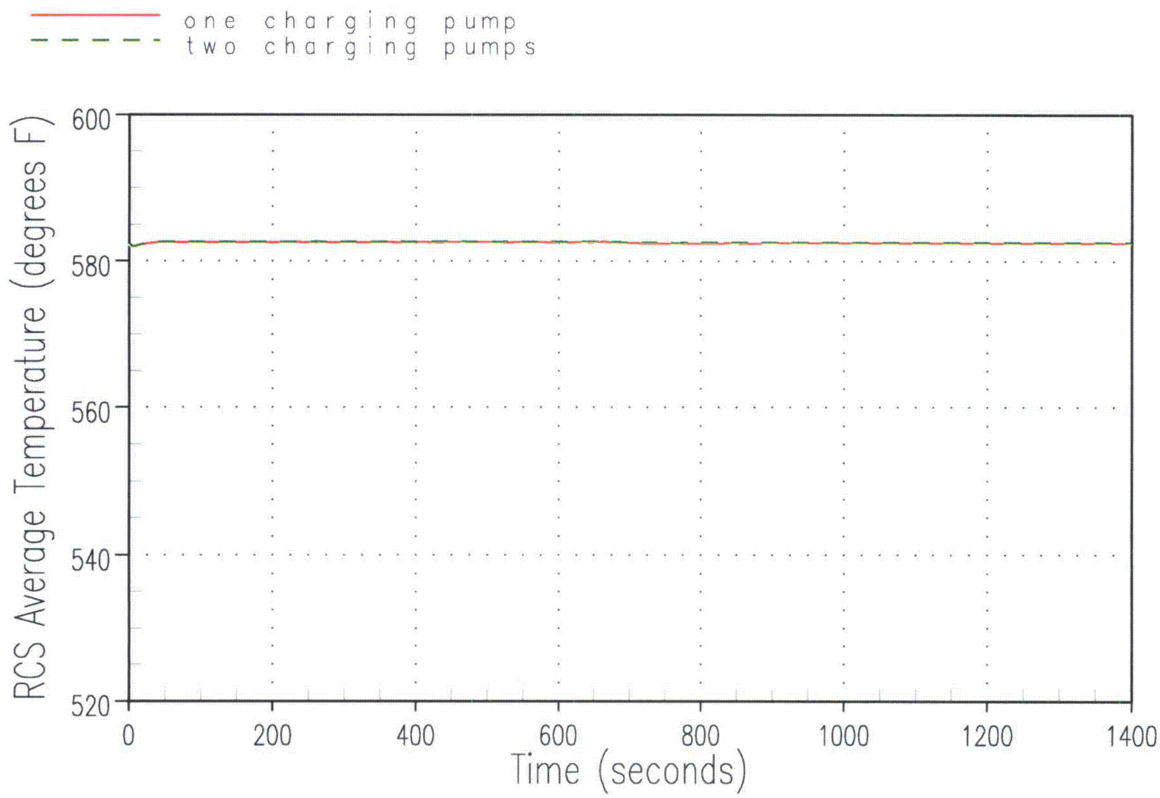
FINAL SAFETY

ANALYSIS REPORT

CVCS MALFUNCTION

NUCLEAR POWER VS. TIME

FIGURE 15.2.15-1



WATTS BAR NUCLEAR PLANT

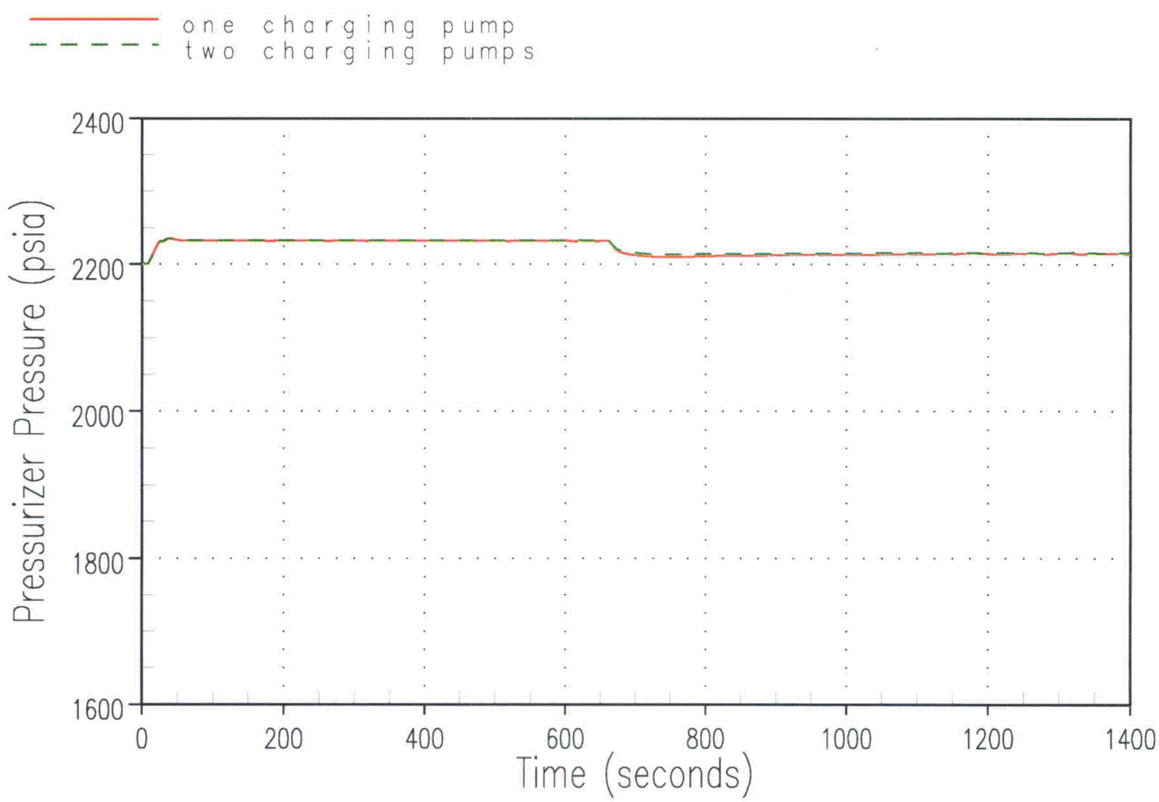
FINAL SAFETY

ANALYSIS REPORT

CVCS MALFUNCTION

RCS AVERAGE TEMPERATURE VERSUS TIME

FIGURE 15.2.15-2



WATTS BAR NUCLEAR PLANT

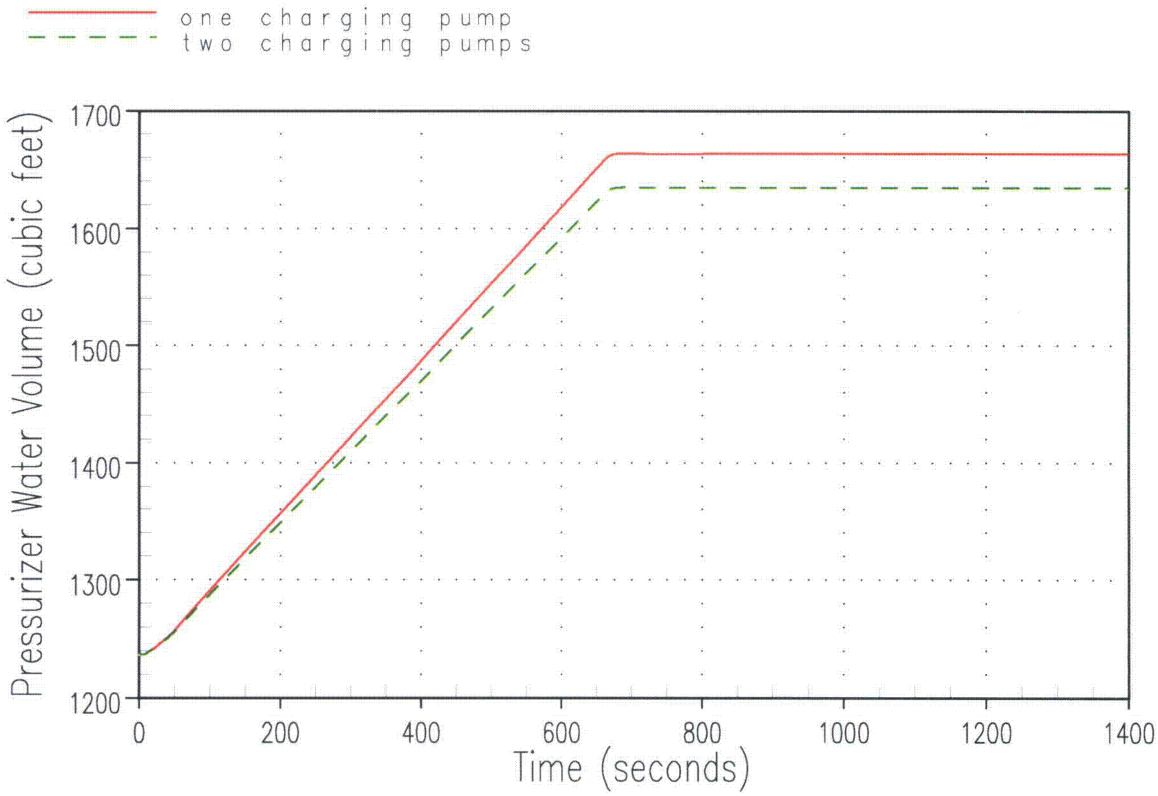
FINAL SAFETY

ANALYSIS REPORT

CVCS MALFUNCTION

PRESSURIZER PRESSURE VERSUS TIME

FIGURE 15.2.15-3



WATTS BAR NUCLEAR PLANT *FINAL SAFETY* *ANALYSIS REPORT*
CVCS MALFUNCTION PRESSURIZER WATER VOLUME VERSUS TIME FIGURE 15.2.15-4