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U.S. Nuclear Regulatory Commission
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ULNRC- 04258

Gentlemen:

DOCKET NUMBER 50-483
UNION ELECTRIC COMPANY
CALLAWAY PLANT
REVISION TO TECHNICAL SPECIFICATIONS 3.3.2, 3.4.10, AND 3.4.11
PRESSURIZER SAFETY VALVES AND PORVS

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References: 1. OL Amendment No. 133 dated May 28, 1999, Improved Technical Specifications
2. ULNRC-03876 dated August 5, 1998 (LER 98-006-00)
3. ULNRC-04205 dated March 21, 2000 (LER 2000-003-00)

Union Electric Company herewith transmits an application for amendment to Facility Operating License No. NPF-30 for the Callaway Plant.

This amendment application would revise the LCO for Technical Specification 3.4.10, "Pressurizer Safety Valves," to change the pressurizer safety valve (PSV) lift setting range from the current "$\geq$ 2460 psig and $\leq$ 2510 psig" to "$\geq$ 2411 psig and $\leq$ 2509 psig." The nominal lift setting is changed from 2485 psig to 2460 psig. This revised LCO provides for an "as-found" tolerance of $\pm$ 2%, whereas the revision to SR 3.4.10.1 requires an as-left setting of 2460 psig $\pm$ 1% following testing. To support these changes, Technical Specification 3.3.2, "ESFAS Instrumentation," and Technical Specification 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," would also be revised to reflect that the automatic actuation circuitry for the PORVs is required for operability since automatic actuation of the valves will now be credited in the mitigation of the Inadvertent ECCS Actuation at Power event analyzed in FSAR Section 15.5.1. As such, new LCO and Surveillance Requirements would be added for the automatic PORV actuation instrumentation.

The Callaway Plant Onsite Review Committee and the Nuclear Safety Review Board have reviewed this amendment application. Attachments 1 through 6 provide the Significant Hazards Evaluation, Environmental

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NRR-060

Consideration, Technical Specification Changes, Draft Technical Specification Bases Changes, Draft FSAR Changes, and PORV Control Schematics, respectively, in support of this amendment request. Attachment 3 mark-ups are based on Reference 1 above. Changes to Technical Specification 3.4.11 effectively retract changes approved in Amendment 133 for DOC 4-02-LS-6 and DOC 4-04-LG, i.e., automatic PORV actuation will now be a required safety function and the associated circuitry will be modified to be fully Class 1E. Attachment 4 mark-ups are provided for information only. Final Bases changes will be implemented under our ITS 5.5.14 Bases Control Program after NRC approval of this amendment application. Attachment 5 mark-ups are also provided for information only to highlight the reanalysis results for the Inadvertent ECCS Actuation at Power event. Attachment 6 illustrates the current non-safety PORV control circuitry which will be modified to be fully Class 1E at Refuel 11 in the spring of 2001.

Commitments associated with this amendment application include resetting the PSV lift settings, upgrading the automatic PORV actuation circuitry to fully Class 1E, and revising Emergency Operating Procedure E-0 to reflect the simulator exercises discussed in Attachment 1. Approval of this amendment application is requested by March 1, 2001. The amendment will be fully implemented prior to entering MODE 3 from MODE 4 (PSV lift setting changes implemented prior to entering LCO 3.4.10 Applicability) during startup from Refuel 11.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10CFR50.92. Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

If you have any questions on this amendment application, please contact us.

Very truly yours,

Alan C. Passwater

Alan C. Passwater
Manager-Corporate Nuclear Services

Attachments:

1 - Significant Hazards Evaluation
2 - Environmental Consideration
3 - Technical Specification Changes
4 - Draft Technical Specification Bases Changes
5 - Draft FSAR Changes
6 - PORV Control Schematics

STATE OF MISSOURI)
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CITY OF ST. LOUIS)

Alan C. Passwater, of lawful age, being first duly sworn upon oath says that he is Manager, Corporate Nuclear Services for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Alan C. Passwater

Alan C. Passwater
Manager, Corporate Nuclear Services

SUBSCRIBED and sworn to before me this 25th day of May, 2000.

Melissa L. Orr

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ATTACHMENT ONE

SIGNIFICANT HAZARDS EVALUATION

SIGNIFICANT HAZARDS EVALUATION

INTRODUCTION

This amendment application would revise the LCO for Technical Specification 3.4.10, "Pressurizer Safety Valves," to change the pressurizer safety valve lift setting range from the current "≥2460 psig and ≤2510 psig" to "≥2411 psig and ≤2509 psig." The nominal lift setting is changed from 2485 psig to 2460 psig. This revised LCO provides for an "as-found" tolerance of ±2%, whereas the revision to SR 3.4.10.1 requires an as-left setting of 2460 psig ±1% following testing. To support these changes, Technical Specification 3.3.2, "ESFAS Instrumentation," and Technical Specification 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," would also be revised to reflect that the automatic actuation circuitry for the PORVs is required for operability since automatic actuation of the valves will now be credited in the mitigation of the Inadvertent ECCS Actuation at Power event analyzed in FSAR Section 15.5.1. As such, new LCO and Surveillance Requirements would be added for the automatic PORV actuation instrumentation.

BACKGROUND

References 2 and 3 of the cover letter documented a recurring concern where pressurizer safety valves (PSVs) were found to be outside the ±1% setpoint tolerance allowed by the Technical Specifications. The failure to meet the ±1% PSV setpoint tolerance has been an industry problem that has resulted in several licensees, including Wolf Creek Generating Station and the Vogtle units, requesting license amendments to relax this tolerance. Westinghouse has performed an evaluation to support this license amendment to allow the PSV nominal setpoint to be lowered to 2460 psig with a ±2% "as-found" tolerance and a ±1% "as-left" tolerance.

Reanalysis of the Inadvertent ECCS Actuation at Power event was performed by Westinghouse in support of this amendment application. The reanalysis credits operator actions from the main control room to open a pressurizer PORV block valve (assumed to initially be closed) and assure the PORV handswitches are in the automatic operation position to allow automatic actuation of at least one PORV on demand. This would prevent water relief through the PSVs.

NRC Generic Issue 70 (GI-70), "Power-Operated Relief Valve and Block Valve Reliability," evaluated the reliability of pressurizer PORVs and block valves and their safety significance in PWR plants. GI-70 identified those safety-related functions that may be performed by the PORVs and also identified potential improvements to the PORVs and block valves. NRC Generic Letter 90-06, dated June 25, 1990, states on page 1 of the cover letter: "On the basis of technical studies for GI-70, the staff requests that to enhance safety, actions identified in Section 3 of Enclosure A be taken by all PWR licensees and CP holders that use or could use PORVs to perform any of the safety-related functions identified in Section 2 of Enclosure A. These actions result from the staff interpretation of safety-related equipment (see 10CFR50.49 and 10 CFR Part 100, Appendix A)."

As documented in ULNRC-2343 dated December 18, 1990, Callaway met the applicable requirements of Generic Letter 90-06 Enclosure A. Enclosure A also proposed a Standard Technical Specification for Westinghouse plants with two PORVs (Attachments A-1 and A-3). OL Amendment 83, dated August 5, 1993, revised Callaway Technical Specification 3/4.4.4 for the PORVs to be consistent with the NRC position presented in Enclosure A, Attachments A-1 and A-3. However, an exception was taken to Attachment A-3 where it stated that the operability of the PORVs is based, in part, on their being able to automatically control reactor coolant system pressure to reduce the challenges to the code safety valves. ULNRC-2343 states on page 4: "Westinghouse standards for transient and accident analyses state that control systems are not assumed to operate unless their operation will cause the results of the transient or accident analysis to be more severe. In keeping with the Westinghouse standards, the Callaway accident analyses do not rely on automatic actuation of the PORVs to prevent overpressurization. Therefore, Callaway Technical Specification 3/4.4.4, "Relief Valves," is adequate to ensure the PORVs are available for manual operation for mitigation of a steam generator tube rupture accident and achieving plant cooldown in accordance with Branch Technical Position RSB 5-1 to Standard Review Plant Section 5.4.7."

Since the reanalysis of the Inadvertent ECCS Actuation at Power event will now credit automatic actuation of at least one PORV, the operability and surveillance requirements for the PORVs provided in Technical Specification 3.4.11 will be revised. These changes will be consistent with Generic Letter 90-06 Enclosure A, Attachment A-3.

The power supplies to the PORVs (BBPCV0455A and BBPCV0456A) and their block valves are Class 1E. The automatic pressure relief signal derived from the pressurizer pressure control system is non-1E. This signal is derived from bistables BB-PB-0455E and BB-PB-0456E, which are currently fed from non-1E separation groups 5 and 6 (see FSAR Figure 7.2-1 sheet 11 in Attachment 6). Since the accident analyses do not rely on the automatic actuation of non-safety related control grade systems or components for accident mitigation, a modification will be implemented during the next refueling outage to upgrade the automatic pressurizer PORV pressure relief circuitry to fully Class 1E.

EVALUATIONS

LOCA and MSLB Related Evaluations

The FSAR Chapter 15 LOCA, Steam Generator Tube Rupture (SGTR), and Main Steamline Break (MSLB) analyses all result in a decreasing RCS pressure and do not challenge the PSV opening pressure. The PSV setpoint is not modeled in the evaluations for LOCA, LOCA hydraulic forces, SGTR, MSLB, Instrumentation and Controls/ Equipment Qualification, LOCA Mass & Energy Releases, and MSLB Mass & Energy Releases. Therefore, none of these evaluations are affected by the proposed change to the PSV nominal setpoint and the allowable setpoint tolerance. Additionally, none of these evaluations credit automatic actuation of the pressurizer PORVs.

Locked RCP Rotor and Loss of External Electrical Load/Turbine Trip Analyses

The limiting maximum allowable PSV setpoint requirements are set by the locked RCP rotor and loss of external electrical load/turbine trip transient analyses. All other non-LOCA transients that credit maximum allowable PSV actuation are much less sensitive to the assumed PSV opening setpoint and are not adversely affected by the proposed change. The locked RCP rotor and loss of external electrical load/turbine trip transient analyses assume the PSVs begin to purge the loop seals at a pressure of 2550 psia. This value is conservatively based on a nominal PSV setpoint of 2500 psia plus a 1% setpoint tolerance and a 1% setpoint shift (due to the presence of the water seal). If the proposed setpoint change is considered, the +2% setpoint tolerance combined with the 1% setpoint shift (due to the loop seal) would result in a PSV opening pressure of about 2549 psia [2460*1.03 + 15]. This value remains bounded by the maximum PSV setpoint assumed in the above limiting overpressurization events.

While the locked RCP rotor and loss of external electrical load/turbine trip transient analyses assume PSV actuation at 2550 psia, steam relief is not assumed to occur until the water seals are cleared 1.15 seconds later. The 1.15 second loop seal purge time was calculated using the methodology of WCAP-12910-P-A and approved by NRC for Callaway in OL Amendment 128 dated October 2, 1998. During this time delay, the pressurizer pressure is assumed to continue increasing at a rate of approximately 100 to 120 psi per second. Therefore, steam relief through the PSVs is not assumed to occur until the pressurizer pressure is approximately 2600 to 2650 psig. The minimum design relief capacity of each PSV is 420,000 lb/hr at a pressure of 2485 psig. At a pressure of 2600 psig, the minimum relief capacity would be in excess of 420,000 lb/hr. However, the safety analyses for overpressurization events conservatively assume a 420,000 lb/hr minimum design relief capacity for the PSVs.

While the plant modification will make the automatic PORV pressure relief circuitry fully Class 1E, this function will only be credited in the analysis of the Inadvertent ECCS Actuation at Power event. No credit for automatic pressurizer PORV actuation is taken in any of the other non-LOCA transients.

Inadvertent ECCS Actuation at Power Analysis

The limiting minimum allowable PSV setpoint requirements are set by the Inadvertent ECCS Actuation at Power event. The assumed PSV opening setpoint is used in establishing the appropriate operator action time requirements needed to assure that the PSVs will not be required to operate while the pressurizer is water solid. A lower PSV opening setpoint will require earlier operator action to preclude water relief through the PSVs.

Westinghouse performed the reanalysis of the Inadvertent ECCS Actuation at Power event in support of this amendment application (see Attachment 5). The current analysis credits operator termination of safety injection within 10 minutes, which is questionable under the current version of Emergency Operating Procedure E-0 at Callaway. The reanalysis credits operator actions from the main control room to terminate normal charging pump (NCP) flow at six (6) minutes into the event and to open a PORV block valve (assumed to initially be closed) and assure the PORV handswitches are in the automatic operation position at nine (9) minutes into the event to allow automatic actuation of at least one PORV on demand. These times include all process and instrumentation delays. These actions would prevent water relief through the PSVs.

Since the FSAR accident analyses do not rely on the automatic actuation of non-safety related control systems or components for accident mitigation, a plant modification will upgrade the automatic PORV actuation circuitry depicted in Attachment 6 to fully Class 1E.

Under the proposed change to the PSV nominal setpoint and the allowable setpoint tolerance, the PSVs could potentially open at pressures as low as 2410 psig (although the revised LCO 3.4.10 limits the lower value to 2411 psig to reflect round-off). This value is the proposed PSV nominal setpoint of 2460 psig minus a 2% setpoint tolerance. The PSV setpoint assumed in the Westinghouse reanalysis of the Inadvertent ECCS Actuation at Power event is based on a nominal value of 2500 psia minus a 3% setpoint tolerance. A PSV nominal setpoint of 2500 psia minus a 3% setpoint tolerance is effectively equivalent to a PSV nominal setpoint of 2460 psig minus a 2% setpoint tolerance. Although the PSVs are not actuated during this transient, the assumed PSV opening setpoint serves as a limit to demonstrate the acceptability of the assumed operator action times. The revised FSAR Figure 15.5-2 in Attachment 5 shows that, if timely operator actions are taken to terminate NCP flow and to assure that at least one PORV is available for automatic pressure relief, water relief through the PSVs is precluded. The proposed reduction in the minimum allowable PSV setpoint does not adversely impact any of the other non-LOCA transients.

Operator Actions to Terminate Inadvertent ECCS Actuation at Power Event

Simulator exercises for the Inadvertent ECCS Actuation at Power event were performed on the Callaway Training Simulator on August 10, 1999 to determine the times required for the control room operators to stop the NCP, unblock the PORVs and assure their availability for automatic pressure relief (i.e., the PORV handswitches are in the automatic operation position), and terminate ECCS flow. In all cases, the NCP was stopped within four (4) minutes, the PORVs were unblocked and available for automatic pressure relief within seven (7) minutes, and ECCS flow was terminated within 21 minutes. For operational and procedural flexibility, the reanalysis for this event reflected in Attachment 5 conservatively credits operator actions to stop the NCP in six (6) minutes and to unblock the PORVs and assure their availability for automatic pressure relief in nine (9) minutes. These times include all process and instrumentation delays.

To support the operator action times assumed in the reanalysis, Emergency Operating Procedure E-0 was revised for use in the simulator runs. Changes were made to the steps as a result of the earlier times required of the operators to terminate the inadvertent ECCS event by opening the PORV block valves and assuring the availability of the PORVs for automatic pressure relief. Because of the potentially adverse consequences of terminating ECCS flow too early during an event where ECCS flow is desired, we chose not to pursue ECCS flow termination earlier within the E-0 procedure. The revised E-0 procedure will be formally issued prior to implementing this proposed license amendment.

The simulator exercises for the Inadvertent ECCS Actuation at Power event were performed using a draft of the proposed E-0 procedure revision. The two crews in the simulator exercises were minimally staffed and aware of the intended scenario. Both crews consisted of three licensed operator personnel. All operating crews will be trained on the revised E-0 procedure during their scheduled requalification training.

Training simulator results and the margin included in the reanalysis provide a high degree of confidence that successful termination of the inadvertent ECCS event by the Callaway operators is achievable prior to water relief through the PSVs. However, in the extremely unlikely event that timely operator actions are not taken prior to water relief through the PSVs, no significant safety concerns exist. This conclusion is based on Westinghouse Nuclear Safety Advisory Letter NSAL-93-013, Supplement 1, dated October 28, 1998. In this NSAL, Westinghouse states:

> "Without appropriate operator action to terminate safety injection flow prior to reaching a water-solid pressurizer condition, the Inadvertent ECCS Actuation at Power event may progress from a Condition II to a more severe Condition III LOCA event. While this occurrence may result in a violation of one of the applicable licensing basis criteria for a Condition II event, it is not considered a significant safety concern. As a LOCA event, discharge of coolant out of the PSRVs [herein referred to as PSVs] and PORVs due to ECCS flow is not significantly adverse relative to other Condition II LOCA events currently analyzed. This is because the pressurizer is located on the hot leg (a hot leg LOCA being less severe than a cold leg LOCA) and because the Inadvertent ECCS Actuation at Power event typically models maximum ECCS flow (to maximize the effects of the initiating event) which is a benefit for LOCA. As such, the Inadvertent ECCS Actuation at Power induced LOCA is bounded by the existing small break LOCA analyses."

Since the consequences of a failure of the operators to open the PORV block valve and assure the PORVs are available for automatic pressure relief are the same as a failure of the operators to terminate ECCS flow, and since these consequences are bounded by the consequences of a small break LOCA, the assumed operator action times are acceptable.

<u>Changes to the Pressurizer PORV Automatic Actuation Logic</u>

In the current design, the PORV opening logic is control grade circuitry that actuates on a one out of one (1/1) logic (see FSAR Figure 7.2-1 sheet 11 in

Attachment 6). A selectable control switch is provided so that either one of two channels can be selected to control each PORV. The current PORV closure logic is safety grade based on a two out of four (2/4) pressurizer pressure low logic (see FSAR Figure 7.6-4 sheets 1-3 in Attachment 6). For the modification to upgrade the automatic actuation circuitry to fully Class 1E, the safety grade closure logic will be used in the design of the new safety grade opening logic. The new 2/4 opening actuation logic will be used to open the PORVs at a pressurizer pressure greater than 2350 psia (the same opening pressure as currently set). The new PORV closure actuation logic will be three out of four (3/4) that actuates when the pressurizer pressure drops 20 psi below the opening setpoint. The PORV opening circuitry will require energization to open the PORV. This design will minimize the potential for spurious opening of the PORVs. However, with the new actuation logic, two failed high channels of pressurizer pressure will result in an inadvertent opening of both PORVs and the PORVs would remain open until remote-manually closed. With the current control logic, while a single channel failing high would open one PORV, the closure logic would close that PORV when pressurizer pressure dropped below approximately 2200 psia. It would take three channels failed high for the PORVs to open and remain open in the current design.

After the implementation of the proposed modification discussed above, there will be added assurance that the PORVs will be capable of providing automatic pressure control and preventing challenges to the PSVs under water solid conditions. However, there is a small impact due to the potential for multiple channel failures to result in an increase in the probability of inadvertently opening both PORVs. With the current control logic, three failed high pressurizer pressure channels (3/4) are required for both PORVs to inadvertently open and remain open. With the new actuation logic, two failed high channels of pressurizer pressure will result in an inadvertent opening of both PORVs and the PORVs would remain open until remote-manually closed. Therefore, it is concluded that there is an increase in the probability that the PORVs will inadvertently open and remain open. However, multiple failures are required for this malfunction and failure modes that result in multiple channels failing high are highly unlikely. Failure modes such as loss of electrical power to the transmitter or a failure of the instrument tubing would result in the affected channel failing low. The new logic will be energize to actuate in order to open the PORVs, further minimizing the potential for inadvertent opening. In addition, these multiple failures that result in the PORVs automatically opening and remaining open do not prevent the operators from remote-manually (from the main control room) closing the PORVs by placing the PORV handswitches in the closed position. Therefore, this increase in the probability that the PORVs will inadvertently open and remain open is considered negligible.

Changes to the Pressurizer PORV Block Valve Automatic Actuation Logic

The current safety grade pressurizer PORV closure logic is also used to close the pressurizer PORV block valves (see FSAR Figure 7.6-4 sheets 1-3 in Attachment 6). From the main control room, the pressurizer PORV block valve handswitches can be in either in the OPEN, AUTO, or CLOSE position. When the handswitch is operated in AUTO, the current 2/4 closure logic automatically closes the pressurizer PORV block valves when pressurizer pressure drops below approximately 2200 psia. For the upgrade, the safety grade pressurizer PORV/ pressurizer PORV block valve closure logic will be used in the design of the safety grade pressurizer PORV opening logic. Since the pressurizer PORV block valves are not designed for continuous cycling, it is undesirable for these valves to automatically open and close with the pressurizer PORVs during an inadvertent ECCS actuation event. Therefore, the AUTO feature for the pressurizer PORV block valves will be eliminated. After the proposed modification is implemented, the pressurizer PORV block valves will be manually opened or closed from the main control room; there will be no automatic block valve actuation. The handswitches on the main control board will be changed from "OPEN-AUTO-CLOSE" to "OPEN-CLOSE".

NUREG-0737 Item II.K.3.1, as discussed in FSAR Section 18.2.17.1.1, requires automatic closure of the pressurizer PORV block valves to protect against a SBLOCA resulting from a stuck-open pressurizer PORV. Union Electric's response is documented in FSAR Section 18.2.17.1.2 and states that the Callaway design includes the capability to automatically isolate the pressurizer PORVs. However, Westinghouse has evaluated the necessity of incorporating an automatic pressurizer PORV isolation system and concluded that such a system should not be required. The Westinghouse evaluation is documented in WCAP-9804 which is listed as Reference 2 in FSAR Section 18.2.18. Based on that evaluation, the automatic pressurizer PORV isolation system does not perform a credited safety function and will, therefore, be eliminated.

Control Systems Evaluation

With a nominal setpoint of 2460 psig and a ±2% setpoint tolerance, the PSVs could potentially open at pressures as low as 2410 psig (rounded up in revised LCO 3.4.10 to 2411 psig). This lower PSV actuation setpoint will reduce the margin between the pressurizer PORV and PSV actuation setpoints from 125 psi to 75 psi. A 75 psi margin is considered adequate and should not challenge the PSVs on Condition I transients (normal condition transients). The PORVs will open prior to the PSVs, even with consideration given to instrument uncertainties in the automatic PORV actuation circuitry.

Credit is taken for PSV operation in the NSSS component design transient limits. The current NSSS design transient limits assume a ±3% tolerance on the PSVs. Therefore, the proposed change does not adversely affect these design transient limits and the current design transient limits remain valid.

Mechanical Components and Systems

The PSVs, in conjunction with the Reactor Trip System, provide overpressure protection for the RCS. The overpressure protection requirement is based on the surge of reactor coolant produced as a result of a transient from full load that produces the largest surge. The PSV setpoint is established to maintain the primary system pressure below 110% of the design pressure. The change in the PSV setpoint and in the tolerance of the setpoint does not change the flow characteristics of the PSVs. The proposed change in the minimum allowable PSV setpoint could result in a transient being terminated at a pressure that is lower than that assumed in the transient's analysis. The primary system pressure boundary is not challenged by the minimum allowable PSV setpoint. Since the maximum allowable PSV setpoint is unaffected by the proposed change (change from round-off only, i.e., the current upper setpoint limit, 2485 +1%, is conservatively rounded down in revised LCO 3.4.10 to give a maximum limit of 2509 psig), the primary system pressure boundary is not challenged by the maximum allowable PSV setpoint.

The Callaway pressurizer surge line layout criteria design calculation uses the PSV setpoint as input for determining the maximum allowable surge line L/D. The nominal setpoint pressure of 2485 psig is used in the calculation. A lower nominal setpoint of 2460 psig is conservative for the results of the calculation. Therefore, the proposed change does not affect the calculation. The Callaway PRT level alarm setpoint calculation also uses the PSV setpoint as an input in determining the PRT alarm setpoints. The nominal setpoint pressure of 2485 psig is used in the calculation. A lower nominal setpoint of 2460 psig is again conservative for the results of the calculation. Therefore, the proposed change does not affect the alarm setpoint calculation.

DISCUSSION OF ASSOCIATED TECHNICAL SPECIFICATION CHANGES

The PSV lift settings in the LCO of Technical Specification 3.4.10, "Pressurizer Safety Valves," are revised from the current "≥2460 psig and ≤2510 psig" to "≥2411 psig and ≤2509 psig." Surveillance Requirement (SR) 3.4.10.1 is revised to specify that the as-left lift setting following testing shall be within ±1% of the new nominal setpoint, 2460 psig.

Conditions A, B, and E of Technical Specification 3.4.11, "Pressurizer PORVs," are revised to reflect differing Required Actions depending on the cause of PORV inoperability, i.e., whether the PORV(s) is inoperable solely due to excessive seat leakage or whether the inoperability is such that the PORV(s) could not be credited to mitigate either a SGTR or an inadvertent ECCS actuation at power event. New LCO and Surveillance Requirements are added to Technical Specification 3.3.2 to reflect the revised operability and testing requirements imposed on the automatic PORV actuation instrumentation. Since this instrumentation is now required to perform a safety function, the same testing and frequency requirements imposed on RTS and ESFAS functions associated with pressurizer pressure in Technical Specification LCOs 3.3.1 and 3.3.2 will now be required of the analog channels and actuation logic used to automatically actuate the PORVs. The Allowable Value was calculated using the Westinghouse setpoint methodology previously discussed in the Bases for LCOs 3.3.1 and 3.3.2. The Allowable Value will not be allowed to increase beyond the nominal trip setpoint by more than the sum of the rack errors (rack calibration accuracy, rack comparator setting accuracy, and rack drift). No response time testing requirements are added since: 1) accident mitigation is dependent on manual operator actions; 2) the assumed operator action times include all process and instrumentation delays; and 3) response time testing requirements for the high pressurizer pressure reactor trip function were eliminated per NRC letter dated March 3, 2000 and the proposed modification will use the same design 7300 Process Protection System and SSPS circuit components. Corresponding Bases changes will also be made.

10CFR50.92 EVALUATION

The proposed change does not involve a significant hazards consideration because operation of Callaway Plant in accordance with this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The pressurizer safety valves (PSVs), in conjunction with the Reactor Trip System (RTS), provide overpressure protection for the Reactor Coolant System (RCS). The PSV setpoint is established to maintain the RCS pressure below 110% of the system design pressure. The proposed change in the minimum allowable PSV setpoint could result in a transient being terminated at a pressure that is lower than that assumed in the transient's analysis. However, the primary system pressure boundary is not challenged by the minimum allowable PSV setpoint. Since the maximum allowable PSV setpoint is unaffected by the proposed change (other than from round-off, as discussed previously), the primary system pressure boundary is not challenged by the maximum allowable PSV setpoint.

With a nominal setpoint of 2460 psig and a ±2% setpoint tolerance, the PSV actuation setpoint could potentially open at pressures as low as 2410 psig (rounded up in revised LCO 3.4.10 to 2411 psig). This lower PSV actuation setpoint will reduce the margin between the pressurizer PORV and PSV actuation setpoint from 125 psi to 75 psi. A 75 psi margin is considered adequate and should not challenge the PSVs on Condition I transients.

The majority of the Callaway PRA event trees question the capability of the PORVs to open for RCS cooldown and depressurization or for feed and bleed cooling. Some event trees question the capability of the PORVs to reclose to terminate RCS depressurization and coolant inventory loss. The transient-induced ATWS event trees question the capability of the PSVs to reclose after opening for these high pressure transients. The maximum allowable PSV setpoint is essentially unchanged; therefore, the proposed change will not adversely impact the probability of the PSVs failing open. Upgrading the automatic PORV actuation circuitry to fully Class 1E, and revising the Technical Specification operability and surveillance requirements to demonstrate the operability of the automatic PORV actuation circuitry, will enhance valve reliability and assure compliance with NRC Generic Letter 90-06. However, it has been determined that this plant modification increases the probability that the PORVs will inadvertently open and remain open if multiple transmitter failures are postulated. With the new safety grade PORV 2/4 opening actuation logic, two failed high pressurizer pressure channels would result in inadvertent opening of both PORVs and the PORVs would remain open until remote-manually closed. Since two of the four channels available to reclose the PORVs are assumed to have failed high, and since closure of the PORVs would require a 3/4 logic to close after the modification is implemented, there would be no signal to close the PORVs on a low pressurizer pressure signal. With the current opening logic, a single failed high pressurizer pressure channel would result in opening one PORV. However, the current 2/4 closure logic would reclose that PORV when pressurizer pressure drops below approximately 2200 psia. With the current control logic, three failed high pressurizer pressure channels (3/4) are required for both PORVs to inadvertently open and remain open. However, the consequences of both PORVs inadvertently opening and remaining open are bounded by the analysis in FSAR Section 15.6.1, "Inadvertent Opening of a Pressurizer Safety or Relief Valve." Since a pressurizer safety valve is sized to relieve approximately twice the steam flow rate of a pressurizer PORV, and will therefore allow a much more rapid depressurization upon opening, the analysis in Section 15.6.1 examines the accidental depressurization of the RCS associated with an inadvertent opening of a pressurizer safety valve. While there is no way to isolate a stuck-open pressurizer safety valve, two open PORVs can be remote-manually isolated by either closing the PORVs or the PORV block valves. Since there is a small impact due to multiple channel failures resulting in an increase in the probability of both PORVs inadvertently opening and remaining open, it is concluded that the proposed activity increases the probability of occurrence of an accident previously evaluated in the FSAR. However, multiple failures are required for this malfunction and failure modes that result in multiple channels failing high are highly unlikely. Therefore, this increase in the probability that the PORVs will inadvertently open and remain open is considered to be insignificant.

All evaluations performed for overpressure transients conservatively assume the upper limit of the PSV tolerance as the pressure to which the RCS is subjected. It has been determined that the design transients are not adversely affected because the limiting transients are not sensitive to the pressure tolerance change. Although the lower PSV setpoint would result in a lower PSV relief flow rate, the slightly lower valve flow rate would be more than compensated for by the reduced valve opening pressure. The change to the PSV setpoint and setpoint tolerance does not change the conclusions of the existing thermal-hydraulic and stress analyses for the pressurizer safety and relief system. The design function of the valves is not being changed and the conclusions documented in the NRC Safety Evaluation of Callaway's response to NUREG-0737 Item II.D.1 (dated September 10, 1987) are unchanged (see also FSAR Section 18.2.5). The PORVs and associated discharge piping can accommodate water relief.

Overall protection system performance will remain within the assumptions of the previously performed accident analyses since the only hardware changes are associated with making the automatic PORV actuation circuitry fully Class 1E. The RTS and Engineered Safety Feature Actuation System (ESFAS) protection systems will continue to function in a manner consistent with the plant design basis. The automatic PORV actuation circuitry modification will be performed in such a manner that all design, material, and construction standards that were applicable to safety-related systems prior to the change are maintained.

The proposed change will not affect the probability of any event initiators nor will the proposed change negatively affect the ability of any safety-related equipment to perform its intended function. Changing the PSV lift setting does not change the probability that an event will occur which will result in the PSV opening. There will be no degradation in the performance of safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters.

Since the FSAR Chapter 15 LOCA, SGTR and MSLB analyses all result in decreasing RCS pressure and do not challenge the PSV opening pressure, none of these events are affected by the proposed change to the PSV nominal setpoint and the allowable setpoint tolerance. Timely operator actions will be taken to preclude water relief through the PSVs during an Inadvertent ECCS Actuation at Power event. Water relief from the PORVs for the latter event would result in a larger discharge of RCS inventory than currently analyzed, wherein operator action is assumed to terminate safety injection within 10 minutes prior to the pressurizer filling. However, FSAR Figure 15.5-3 in Attachment 5 demonstrates that DNB is not a concern, there will be no fuel failures associated with this event, and RCS inventory will be directed to the pressurizer relief tank located inside containment. Therefore, there will no impact on offsite radiological consequences. None of the other non-LOCA transients are adversely affected by the proposed change. Since none of the other FSAR Chapter 15 events are adversely affected, the radiological consequences of those events are not adversely affected.

In the Westinghouse reanalysis of the Inadvertent ECCS Actuation at Power event, the minimum PSV opening setpoint serves as a limit to demonstrate the acceptability of the assumed operator action times to assure that the PSVs will not be required to operate while the pressurizer is water solid. A lower PSV opening setpoint could potentially require earlier operator actions to prevent water relief through the PSVs. Simulator exercises for the Inadvertent ECCS Actuation at Power event were performed on the Callaway training simulator on August 10, 1999 to determine the times required for the control room operators to stop the NCP and unblock the PORVs and assure their availability for automatic pressure relief. In all cases, the NCP was stopped within four (4) minutes and the PORVs were unblocked and available for automatic pressure relief within seven (7) minutes. The reanalysis in Attachment 5 conservatively credits operator actions from the main control room to stop the NCP in six (6) minutes and to unblock the PORVs and assure their availability for automatic pressure relief in nine (9) minutes. These times include all process and instrumentation delays. The revised FSAR Figure 15.5-2 shows that if operator actions are taken within these time frames to terminate NCP flow and to assure at least one PORV is available for automatic pressure relief, water relief through the PSVs is precluded. Procedure changes and periodic operator requalification training will provide assurance that these operator actions can be performed within the assumed time constraints.

Based on the above discussions, the proposed change will not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

The nominal setpoint for the PSVs will be lowered by 1% from 2485 psig to 2460 psig. The allowable setpoint tolerance will be increased from ±1% to ±2%. The combined effect of these changes results in a 2% decrease in the minimum acceptable PSV setpoint from 2460 psig to 2411 psig. The change in the PSV setpoint and in the tolerance of the setpoint does not change their ability to open on demand. The maximum acceptable PSV setpoint is unaffected by this proposed change, other than round-off as discussed previously. Since the FSAR accident analyses do not rely on the automatic actuation of non-safety related control grade systems or components for accident mitigation, a plant modification will make the automatic pressurizer PORV pressure relief circuitry fully Class 1E.

The proposed change to the PSV nominal setpoint and the allowable setpoint tolerance will not prevent the PSVs from performing their RCS overpressurization protection function. Additionally, the proposed change does not affect the ability of any other safety-related equipment to perform its safety function.

The only hardware changes are associated with making the automatic PORV actuation circuitry fully Class 1E. The RTS and Engineered Safety Feature Actuation System (ESFAS) protection systems will continue to function in a manner consistent with the plant design basis. The automatic PORV actuation circuitry modification will be performed in such a manner that all design, material, and construction standards that were applicable to safety-related systems prior to the change are maintained. While the possibility that the PORVs fail to control RCS pressure, that at least one PORV fails to open, and that the operator fails to open the block valve and assure the PORV(s) are available for automatic pressure relief within the required time frame are all malfunctions of a different type than currently analyzed in the FSAR, they do not create different accident types. The Class 1E upgrade and changes to Emergency Operating Procedure E-0 will provide assurance that the reanalysis presented in Attachment 5 will bound the results of this event which, in turn, is also bounded by the results presented in FSAR Section 15.6.1 for an inadvertent PSV opening.

There are no other changes in the method by which any safety-related plant system performs its safety function. The change will not affect the normal method of plant operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Involve a significant reduction in a margin of safety.

The PSVs, in conjunction with the RTS, provide overpressure protection for the RCS. The change in the upper limit of the PSV tolerance from ±1% to ±2%, with a reduction in the nominal setpoint from 2485 psig to 2460 psig, does not challenge the upper limit of overpressure protection. The maximum opening pressure setpoint is unchanged (other than a conservative round-off), and therefore, does not impact analyses performed for overpressure transients. The change to the PSV setpoint and setpoint tolerance does not change the conclusions of the existing thermal-hydraulic and stress analyses for the pressurizer safety and relief system. For all non-LOCA events, the above evaluations support the change in the PSV setpoint and setpoint tolerance from 2485 psig ±1% to 2460 psig ±2%. The change in the PSV setpoint and setpoint tolerance also has no effect on the RTS or ESFAS trip setpoints.

The Bases for Technical Specification 3.4.10 states the following in the Background section:

"The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure...The relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer...."

The locked RCP rotor and loss of external electrical load/turbine trip transient analyses assume PSV actuation at 2550 psia. This value is conservatively based on a nominal PSV setpoint of 2500 psia plus a 1% setpoint tolerance and a 1% setpoint shift (due to the presence of the water seal). The maximum allowable PSV setpoint of 2509 psig is unaffected by the proposed change, other than a conservative round-off discussed previously. At a pressure of 2509 psig, the minimum relief capacity of the safety valves would be in excess of 420,000 lb/hr. However, the safety analyses for overpressurization events conservatively assume a 420,000 lb/hr minimum design relief capacity for the PSVs.

The proposed change does not affect the acceptance criteria for any other analyzed event nor is there a change to any other Safety Analysis Limit (SAL). The acceptance criteria for the Inadvertent ECCS Actuation at Power event will remain the same as currently analyzed; however, operator action and automatic PORV actuation will be relied upon to demonstrate compliance with that event's acceptance criteria.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR limits, F_Q, $F\Delta H$, LOCA PCT, peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan continue to be met.

Therefore, the proposed change does not involve a significant reduction in any margin of safety.

CONCLUSION

Based upon the preceding information, it has been determined that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10CFR50.92(c) and does not involve a significant hazards consideration.

ATTACHMENT TWO

ENVIRONMENTAL CONSIDERATION

ENVIRONMENTAL CONSIDERATION

This amendment application would revise the LCO for Technical Specification 3.4.10, "Pressurizer Safety Valves," to change the pressurizer safety valve lift setting range from the current "≥ 2460 psig and ≤ 2510 psig" to "≥ 2411 psig and ≤ 2509 psig." The nominal lift setting is changed from 2485 psig to 2460 psig. This revised LCO provides for an "as-found" tolerance of ± 2%, whereas the revision to SR 3.4.10.1 requires an as-left setting of 2460 psig ± 1% following testing. To support these changes, Technical Specification 3.3.2, "ESFAS Instrumentation," and Technical Specification 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," would also be revised to reflect that the automatic actuation circuitry for the PORVs is required for operability since automatic actuation of the valves will now be credited in the mitigation of the Inadvertent ECCS Actuation at Power event analyzed in FSAR Section 15.5.1. As such, new LCO and Surveillance Requirements would be added for the automatic PORV actuation instrumentation.

The proposed amendment involves changes with respect to the use of facility components located within the restricted area, as defined in 10CFR20. Union Electric has determined that the proposed amendment does not involve:

(1) A significant hazards consideration, as discussed in Attachment 1 of this amendment application;

(2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite;

(3) A significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR51.22(c)(9). Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

ATTACHMENT THREE

TECHNICAL SPECIFICATION CHANGES

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One channel or train inoperable.	F.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	F.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	F.2.2 Be in MODE 4.	60 hours
G. One train inoperable.	------------------- NOTE ------------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. --	
	G.1 Restore train to OPERABLE status.	6 hours
	<u>OR</u>	
	G.2.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	G.2.2 Be in MODE 4.	18 hours
H. ~~Not used.~~		

(continued)

INSERT 3.3-28

INSERT 3.3-28

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. One or more trains inoperable.	-------------------- NOTE ------------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. -- H.1 Declare associated Pressurizer PORV(s) inoperable.	 Immediately

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.13	-------------------------------NOTE------------------------------ Only applicable to slave relays K602, K622, K624, K630, K740, and K741. -- Perform SLAVE RELAY TEST.	18 months <u>AND</u> Prior to entering MODE 4 when in MODE 5 or 6 > 24 hours, if not performed within the previous 92 days
SR 3.3.2.14	-------------------------------NOTE------------------------------ Only applicable to slave relays K620 and K750. -- Perform SLAVE RELAY TEST.	18 months <u>AND</u> Prior to entering MODE 3 when in MODE 5 or 6 > 24 hours, if not performed within the previous 92 days

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater (continued)					
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	O	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.12	≥ 20.64 psia
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.

INSERT 3.3-44

INSERT 3.3-44

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure - High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤ 2350 psig

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings $\geq$ ~~2460~~ 2411 psig and $\leq$ ~~2510~~ 2509 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 275° F.

--NOTE--
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperature $\leq$ 275° F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1% of 2460 psig.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTES---

1. Separate Condition entry is allowed for each PORV.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more PORVs inoperable ~~and capable of being manually cycled.~~ solely due to excessive seat leakage.	A.1	Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable ~~and not capable of being manually cycled.~~ for reasons other than excessive seat leakage.	B.1	Close associated block valve.	1 hour
	<u>AND</u>		
	B.2	Remove power from associated block valve.	1 hour
	<u>AND</u>		
	B.3	Restore PORV to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	------------------- NOTE ------------------- Required Actions do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2. --		
	C.1	Place associated PORV in manual control.	1 hour
	AND		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable ~~and not capable of being manually cycled.~~ for reasons other than excessive seat leakage.	E.1	Close associated block valves.	1 hour
	AND		
	E.2	Remove power from associated block valves.	1 hour
	AND		
	E.3	Be in MODE 3.	6 hours
	AND		
	E.4	Be in MODE 4.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. One channel or train inoperable.	F.1	Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>		
	F.2.1	Be in MODE 3.	54 hours
	<u>AND</u>		
	F.2.2	Be in MODE 4.	60 hours
G. One train inoperable.	-------------------- NOTE ------------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---		
	G.1	Restore train to OPERABLE status.	6 hours
	<u>OR</u>		
	G.2.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	G.2.2	Be in MODE 4.	18 hours
H. One or more trains inoperable.	-------------------- NOTE ------------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---		
	H.1	Declare associated Pressurizer PORV(s) inoperable.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.13	------------------------------NOTE------------------------------ Only applicable to slave relays K602, K622, K624, K630, K740, and K741. --- Perform SLAVE RELAY TEST.	18 months <u>AND</u> Prior to entering MODE 4 when in MODE 5 or 6 > 24 hours, if not performed within the previous 92 days
SR 3.3.2.14	------------------------------NOTE------------------------------ Only applicable to slave relays K620 and K750. --- Perform SLAVE RELAY TEST.	18 months <u>AND</u> Prior to entering MODE 3 when in MODE 5 or 6 > 24 hours, if not performed within the previous 92 days

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[(a)]
6. Auxiliary Feedwater (continued)					
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	O	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.12	≥ 20.64 psia
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure – High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤2350 psig

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings $\geq$ 2411 psig and $\leq$ 2509 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 275°F.

--- NOTE ---
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with any RCS cold leg temperature $\leq$ 275°F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1% of 2460 psig.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each PORV.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable solely due to excessive seat leakage.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable for reasons other than excessive seat leakage.	B.1 Close associated block valve.	1 hour
	AND	
	B.2 Remove power from associated block valve.	1 hour
	AND	
	B.3 Restore PORV to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	------------------- NOTE ------------------ Required Actions do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2. ---		
	C.1	Place associated PORV in manual control.	1 hour
	<u>AND</u>		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable for reasons other than excessive seat leakage.	E.1	Close associated block valves.	1 hour
	<u>AND</u>		
	E.2	Remove power from associated block valves.	1 hour
	<u>AND</u>		
	E.3	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.4	Be in MODE 4.	12 hours

(continued)

ATTACHMENT FOUR

DRAFT TECHNICAL SPECIFICATION BASES CHANGES

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

c. Safety Injection - Containment Pressure - High 1 (continued)

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties. The Trip Setpoint is ≤ 3.5 psig.

Containment Pressure - High 1 must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure - Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) atmospheric steam dump valve or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer PORV or safety valve;
- LOCAs; and
- SG Tube Rupture.

The pressurizer pressure channels provide both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, ~~and~~ SI, and automatic PORV actuation. Therefore, the actuation logic must be able to withstand both an input failure to control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

b. Engineered Safety Feature Actuation System Interlocks - Pressurizer Pressure, P-11 (continued)

disabled. The Trip Setpoint reflects only steady state instrument uncertainties. The Trip Setpoint is ≤ 1970 psig.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

INSERT →
B 3.3.2-36

The ESFAS instrumentation satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified on a per steam line, per SG, per pump, etc., basis, then the Condition may be entered separately for each steam line, SG, pump, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

(continued)

9. Automatic Pressurizer PORV Actuation

For the inadvertent ECCS actuation at power event (a Condition II event), the safety analysis (Ref. 15) credits operator actions from the main control room to terminate flow from the normal charging pump (NCP) and to open at least one PORV block valve (assumed to initially be closed) and assure the availability of the PORV for automatic pressure relief. Analysis results indicate that water relief through the pressurizer safety valves, which could result in the Condition II event degrading into a Condition III event if the safety valves did not reseat, is precluded if operator actions are taken within the times assumed in the Reference 15 analysis to terminate NCP flow and to assure at least one PORV is available for automatic pressure relief. The assumed operator action times conservatively bound the times measured during simulator exercises. Therefore, automatic PORV operation is an assumed safety function in MODES 1, 2, and 3. The PORVs are equipped with automatic actuation circuitry and manual control capability. The PORVs are considered OPERABLE in either the automatic or manual mode, as long as the automatic actuation circuitry is OPERABLE and the PORVs can be made available for automatic pressure relief by timely operator actions (Ref. 15) to open the associated block valves (if closed) and to assure the PORV handswitches are in the automatic operation position. The automatic mode is the preferred configuration, as this provides the required pressure relieving capability without reliance on operator actions.

a. Automatic Pressurizer PORV Actuation - Automatic Actuation Logic and Actuation Relays (SSPS)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for Function 1.b, except that the LCO is not applicable in MODE 4 as discussed below for Function 9.b.

b. Automatic Pressurizer PORV Actuation - Pressurizer Pressure - High

This signal provides protection against an inadvertent ECCS actuation at power event. Pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, SI, and automatic PORV actuation. Therefore, the actuation logic must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four opening logic. The Trip Setpoint is $\leq$ 2335 psig.

The automatic PORV opening logic is satisfied when two-out-of-four (2/4) pressurizer pressure channels exceed their setpoint. Continued operation is allowed with one inoperable channel in the tripped condition. In this case, the automatic opening logic would revert to one-out-of-three (1/3). A single failure (e.g., failed bistable card) in one of the remaining three channels could result in both PORVs opening and remaining open since the automatic closure logic requires three-out-of-four (3/4) channels to reset, which could not be satisfied with two inoperable channels. However, this event can be terminated by PORV block valve closure and the consequences of this event are bounded by the analysis of a stuck open pressurizer safety valve in Reference 16. Therefore, automatic PORV closure is not a required safety function and the OPERABILITY requirements are satisfied by four OPERABLE pressurizer pressure channels.

Consistent with the Applicability of LCO 3.4.11, "Pressurizer PORVs," the LCO for Function 9 is not applicable in MODE 4 when both pressure and core energy are decreased and transients that could cause an overpressure condition will be slow to occur. This is also consistent with the Applicability of Functions 1.c, 1.d, and 1.e. LCO 3.4.12 addresses automatic PORV actuation instrumentation requirements in MODES 4 (with any RCS cold leg temperature $\leq$ 275°F), 5, and 6 with the reactor vessel head in place.

BASES

ACTIONS

C.1, C.2, C.3.1, and C.3.2 (continued)

This action addresses the train orientation of the SSPS and the master and slave relays. Containment Isolation Phase A is the primary signal to ensure closing of the containment purge supply and exhaust valves. If one Phase A train is inoperable, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust valves in their closed position is met. Required Action C.1 is modified by a Note that this Action is only required if Containment Phase A Isolation (Function 3.a.(2)) is inoperable. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of Reference 8 that 4 hours is the average time required to perform channel surveillance.

D.1, D.2.1, and D.2.2

Condition D applies to:

- Containment Pressure - High 1;
- Pressurizer Pressure - Low;
- Steam Line Pressure - Low;
- Containment Pressure - High 2;
- Steam Line Pressure - Negative Rate - High;
- SG Water Level - Low Low (Adverse Containment Environment); ~~and~~

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

- SG Water Level - Low Low (Normal Containment Environment); and
- Pressurizer Pressure - High.

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic (excluding Pressurizer Pressure - Low and SG Water Level - Low Low (Adverse and Normal Containment Environment)). Therefore, failure of one channel (i.e., with the bistable not tripped) places the Function in a two-out-of-two configuration. The inoperable channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

, Pressurizer Pressure - High,

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 8.

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray Containment Pressure - High 3; and
- Containment Phase B Isolation Containment Pressure - High 3.

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore,

(continued)

BASES

ACTIONS

F.1, F.2.1, and F.2.2 (continued)

OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.

G.1, G.2.1, and G.2.2

Condition G applies to the automatic actuation logic and actuation relays (SSPS) for the Steam Line Isolation, Turbine Trip and Feedwater Isolation, and AFW actuation Functions. Condition G also applies to the MSFIS automatic actuation logic.

The action addresses the train orientation of the actuation logic for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Refs. 8 and 13) assumption that 4 hours is the average time required to perform channel surveillance.

~~H.1 and H.2~~

~~Not used.~~

INSERT B 3.3.2-41

(continued)

INSERT B 3.3.2-41

H.1

Condition H applies to the automatic actuation logic and actuation relays (SSPS) for the Automatic Pressurizer PORV Actuation Function.

The Required Action addresses the impact on the ability to mitigate an inadvertent ECCS actuation at power event that requires the availability of at least one pressurizer PORV for automatic pressure relief. With one or more automatic actuation logic trains inoperable, the associated pressurizer PORV(s) must be declared inoperable immediately. This requires that Condition B or E of LCO 3.4.11, "Pressurizer PORVs," be entered immediately depending on the number of PORVs inoperable

The Required Action is modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Refs. 8 and 13) assumption that 4 hours is the average time required to perform channel surveillance.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.8 (continued)

experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of Reference 6.

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable. This does not include verification of time delay relays. These are verified via response time testing per SR 3.3.2.10.

Whenever an RTD is replaced in Function 5.e.(3) or 6.d.(3), the next required CHANNEL CALIBRATION of the RTDs is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.
The portion of the automatic PORV actuation circuitry required for COMS is calibrated in accordance with SR 3.4.12.9.

SR 3.3.2.10

This SR verifies the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response time verification acceptance criteria are included in Reference 9. No credit was taken in the safety analyses for those channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Reference 9. Individual component response times are not modeled in the analyses. The

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.14

SR 3.3.2.14 is the performance of a SLAVE RELAY TEST as described in SR 3.3.2.6, except that SR 3.3.2.14 has a Note specifying that it applies only to slave relays K620 and K750. These ~~This~~ slave relays are ~~is~~ tested with a Frequency of 18 months and prior to entering MODE 3 for Functions 5.a and 9.a whenever the unit has been in MODE 5 or 6 for > 24 hours, if not performed within the previous 92 days. ~~(Reference 12).~~ The 18 month Frequency for ~~this~~ these slave relays is based on the need to perform this Surveillance under the conditions that apply during a unit outage to avoid the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The SLAVE RELAY TEST of relay K620 does not include the circuitry associated with the main feedwater pump trip solenoids since that circuitry serves no required safety function. INSERT B 3.3.2-54

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. Callaway Setpoint Methodology Report (NSSS), SNP (UE)-565 dated May 1, 1984, and Callaway Instrument Loop Uncertainty Estimates (BOP), J-U-GEN.
7. Not used.
8. Callaway OL Amendment No. 64 dated October 9, 1991.
9. FSAR Section 16.3, Table 16.3-2.
10. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.
11. Callaway OL Amendment No. 43 dated April 14, 1989.
12. SLNRC 84-0038 dated February 27, 1984.
13. Callaway OL Amendment No. 117 dated October 1, 1996.
14. WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.
15. FSAR, Section 15.5.1.

16. FSAR, Section 15.6.1.
17. Letter from Mel Gray (NRC) to Garry L. Randolph (UE), "Revision 20 of the Inservice Testing Program for Callaway Plant, Unit 1 (TAC No. MA4469)," dated March 19, 1999.

INSERT B 3.3.2-54

The 18 month Frequency for slave relay K620 was accepted by NRC at initial plant licensing based on Reference 12. The 18 month Frequency for slave relay K750 is consistent with that of SR 3.4.11.2 in LCO 3.4.11, "Pressurizer PORVs," which in turn is based on the NRC-approved Inservice Test (IST) program relief request BB-10 on the pressurizer PORVs (Ref. 17). Testing slave relay K750 at power would result in opening the PORVs and depressurizing the RCS. If the PORV block valves are closed, there is not enough pressure to open the PORVs.

Table B 3.3.2-1
(Page 5 of 5)

FUNCTION		NOMINAL TRIP SETPOINT [a]
7.	Automatic Switchover to Containment Sump	
	a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
	b. Refueling Water Storage Tank (RWST) Level - Low Low	≥ 36%
	Coincident with Safety Injection	See Function 1 (Safety Injection).
8.	ESFAS Interlocks	
	a. Reactor Trip, P-4	N.A.
	b. Pressurizer Pressure, P-11	≤ 1970 psig
9.	Automatic Pressurizer PORV Actuation	
	a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.
	b. Pressurizer Pressure - High	≤ 2335 psig

(a) The inequality sign only indicates conservative direction. The as-left value will be within a two-sided calibration tolerance band on either side of the nominal value.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are of the pop type. The valves are spring loaded and self actuated by direct fluid pressure with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are self actuating, they are considered independent components. The minimum relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally between the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 with the reactor vessel head on; however, in MODE 4 with one or more RCS cold leg temperatures ≤ 275°F, MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on the ~~± 1%~~ tolerance requirements assumed in the safety analyses. ~~(Ref. 1) for lifting pressures above 1000 psig.~~ The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

BASES

BACKGROUND (continued)

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal at full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load/turbine trip;

d. Loss of normal feedwater;

e. Loss of non-emergency AC power to station auxiliaries;

f. Locked rotor;

g. Feedwater line break; and

h. Rod cluster control assembly ejection.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation occurs in the FSAR Chapter 15 analysis of events c, f, and g (above) and may be required for any of the above events to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The three pressurizer safety valves are set to open at 2460 psig (slightly below the RCS design pressure, 2485 psig), and within the ~~ASME~~ specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses' assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ~~± 1%~~ tolerance requirements of ~~(Ref. 1) for lifting pressures above 1000 psig.~~ assumed in the safety analyses.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperature ≤ 275°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below 275°F, overpressure protection is provided by the COMS. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ±2% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Chapter 15.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are safety-related DC solenoid operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive seat leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of surveillances on the block valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

automatic pressure relief signal actuation circuitry, the
The power supplies to the PORVs and their block valves are Class 1E. The manual controls, and ~~COMS~~ portion of the actuation circuitry are also Class 1E. ~~The automatic pressure relief signal derived from the pressurizer pressure control system is non-1E and must be isolated, as shown in Reference 1.~~ The PORVs and their associated block valves are powered from two separate safety trains (Ref. 2).
the cold overpressure mitigation system (COMS)

The plant has two PORVs, each having a relief capacity of 210,000lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure -High reactor trip setpoint for all design transients up to and including the design step load decrease with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for cold

(continued)

BASES

BACKGROUND (continued)

overpressure mitigation. See LCO 3.4.12, "Cold Overpressure Mitigation System (COMS)."

APPLICABLE SAFETY ANALYSES

Plant operators may employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

INSERT 1

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR), pressurizer volume, or hot leg saturation criteria are examined (Ref. 3). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint. The DNBR calculation is more conservative, the pressurizer water volume is maximized, and the hot leg saturation temperature is reduced for those transients assuming PORV operation. Events that assume this condition include turbine trip, loss of normal feedwater, loss of non-emergency AC power to station auxiliaries, and the feedline break case with no SI (Ref. 3). ~~The PORVs are equipped with automatic actuation circuitry and manual control capability. No credit for~~ Automatic operation is ~~taken~~ assumed in the Reference 3 analyses ~~for MODE 1, 2, and 3 transients where~~, but operation of the PORVs has a ~~beneficial~~ detrimental impact on the results of the analysis. ~~Therefore, automatic PORV operation is not an assumed safety function in MODES 1, 2, and 3. The PORVs are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.~~

Pressurizer PORVs satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

INSERT 2

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. INSERT 3 ~~An OPERABLE block~~

(continued)

INSERT 1

For the inadvertent ECCS actuation at power event (a Condition II event), the safety analysis (Ref. 1) credits operator actions from the main control room to terminate flow from the normal charging pump (NCP) and to open a PORV block valve (assumed to initially be closed) and assure the availability of at least one PORV for automatic pressure relief. Analysis results indicate that water relief through the pressurizer safety valves, which could result in the Condition II event degrading into a Condition III event if the safety valves did not reseat, is precluded if operator actions are taken within the times assumed in the Reference 1 analysis to terminate NCP flow and to assure at least one PORV is available for automatic pressure relief. The assumed operator action times conservatively bound the times measured during simulator exercises. Therefore, automatic PORV operation is an assumed safety function in MODES 1, 2, and 3. The PORVs are equipped with automatic actuation circuitry and manual control capability. The PORVs are considered OPERABLE in either the automatic or manual mode, as long as the automatic actuation circuitry is OPERABLE and the PORVs can be made available for automatic pressure relief by timely operator actions (Ref. 1) to open the associated block valves (if closed) and assure the PORV handswitches are in the automatic operation position. The automatic mode is the preferred configuration, as this provides the required pressure relieving capability without reliance on operator actions.

INSERT 2

The LCO also requires the PORVs and their automatic actuation circuitry to be OPERABLE, in conjunction with the capability to manually open their associated block valves and assure the availability of the PORVs for automatic pressure relief, to mitigate the effects associated with an inadvertent ECCS actuation at power event. The PORVs are considered OPERABLE in either the automatic or manual mode, as long as the automatic actuation circuitry is OPERABLE and the PORVs can be made available for automatic pressure relief by timely operator actions (Ref. 1) to open the associated block valves (if closed) and assure the PORV handswitches are in the automatic operation position. The automatic mode is the preferred configuration, as this provides the required pressure relieving capability without reliance on operator actions.

INSERT 3

An OPERABLE block valve may be either open and energized, or closed and energized, with the capability to be cycled, since the required safety functions of the block valve are accomplished by manual operation to cycle the block valve. Although typically open to allow PORV operation, the block valve may be OPERABLE when closed to isolate the flow path of an inoperable PORV because of excessive seat leakage. Isolation of an OPERABLE PORV does not render that PORV or block valve inoperable, provided the automatic pressure relief function remains available with timely operator actions (Ref. 1) to open the associated block valve, if closed, and assure the PORV's handswitch is in the automatic operation position. Satisfying the LCO helps minimize challenges to fission product barriers and precludes water relief through the pressurizer safety valves.

BASES

LCO (continued)

~~valve may be either open, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (e.g., as in the case of a PORV with excessive seat leakage). Isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action. Satisfying the LCO helps minimize challenges to fission product barriers.~~

An OPERABLE PORV ~~is required to be capable of manually opening and closing, and not~~ must not be experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criterion, exists when conditions dictate closure of the block valve to limit leakage.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. ~~Although not a required function, the PORVs' OPERABILITY in MODES 1, 2, and 3 also serves the desired function of minimizing challenges to the pressurizer safety valves.~~ INSERT 4 The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 for manual actuation to mitigate a steam generator tube rupture event.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for COMS in MODES 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits MODE changes with inoperable PORVs or block valves as one possible recourse to remaining in the Applicability of LCO 3.4.12.

(continued)

INSERT 4

The PORVs are required to be OPERABLE in MODES 1, 2, and 3 for automatic pressure relief to fulfill the required function of minimizing challenges to the pressurizer safety valves during an inadvertent ECCS actuation event.

because of excessive seat leakage

BASES

ACTIONS (continued)

automatic pressure relief and capable of

A.1

The PORVs may be inoperable yet capable of being manually cycled. ~~(e.g., excessive seat leakage).~~ In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valves must be closed, but power must be maintained to the associated block valves, since removal of power would render the block valve

INSERT 5

inoperable. ~~No credit for automatic PORV operation is taken in the Reference 3 analyses for MODE 1, 2, and 3 transients. As such the PORVs are considered OPERABLE in either the manual or automatic mode. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to excessive seat leakage or other causes that do not prevent manual use and do not create a possibility for a small break LOCA.~~ Closure of the block valve(s) establishes reactor coolant pressure boundary (RCPB) integrity for a PORV(s) with excessive seat leakage. RCPB integrity takes priority over the capability of the PORV(s) to mitigate an overpressure event. For these reasons, the block valve may be closed but the Action requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE and automatic actuation status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

for reasons other than excessive seat leakage

B.1, B.2, and B.3

If one PORV is inoperable (i.e., not capable of being manually cycled), it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable ~~valve~~ cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the

for Required Actions B.1 and B.2

PORV

automatic pressure relief or not capable of

(continued)

INSERT 5

Credit for automatic PORV operation is taken in the Reference 1 safety analysis. However, the PORVs are considered OPERABLE in either the manual or automatic mode, as long as the automatic actuation circuitry is OPERABLE and the PORV can be made available for automatic pressure relief by timely operator actions (Ref. 1). Although a PORV may be designated inoperable, it may be available for automatic pressure relief and capable of being manually opened and closed and, therefore, able to perform its required safety functions. PORV inoperability solely due to excessive seat leakage does not prevent automatic and manual use and does not create a possibility for a small break LOCA.

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

PORV cannot be restored within this additional time, the plant must be brought to MODE 4, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not fully open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be restored within this additional time, the plant must be brought to MODE 4, as required by Condition D.

The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

automatic pressure relief or not capable of

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 1,2,3, 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 & for requirements in MODES 4,5, and 6.

E.1, E.2, E.3, and E.4

If more than one PORV is inoperable for reasons other than excessive seat leakage, ~~and not capable of being manually cycled,~~ it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one PORV is restored and one PORV remains inoperable, then the plant will be in Condition B with the time clock started at the time the remaining PORV was discovered to be inoperable. If no PORVs are restored within the Completion Time, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 1,2,3, 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 & for requirements in MODES 4,5, and 6.

F.1 and F.2

If more than one block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

The Required Actions are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

automatic pressure relief or not capable of

G.1 and G.2

If the Required Actions of Condition F are not met, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 (with any RCS cold leg temperature ≤ 275°F), 5, and 6 (with the reactor vessel head on), automatic PORV OPERABILITY is required. See LCO 3.4.12 for requirements in MODES 4, 5, and 6.

1, 2, 3,

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed. The basis for the Frequency of 92 days is the ASME Code, Section XI (Ref. 4).

The Note modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Actions of this LCO. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. Operating experience has shown that these valves usually pass the Surveillance when performed at the required Inservice Testing Program frequency. The Frequency is acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. FSAR ~~Figures 7.2-1 (sheet 11) and 7.6-4 (sheets 1-3).~~, Section 15.5.1.
2. Regulatory Guide 1.32, February 1977.
3. FSAR, Section 15.2.
4. ASME, Boiler and Pressure Vessel Code, Section XI.

ATTACHMENT FIVE

DRAFT FSAR CHANGES

to that which results from a single sprung flange, a single pump seal failure, a single valve stem packing failure, or other single failure mechanisms considered credible by a systematic analysis of system components. The probability of a large break in a piping system (e.g., rupture of ECCS piping), subsequent to the original large LOCA pipe break, is considered to be sufficiently low that it need not be postulated.

Single failures of passive components in electrical systems are assumed in designing against a single failure.

3.1.2 ADDITIONAL SINGLE FAILURE ASSUMPTIONS

In designing for and analyzing for a DBA (i.e., loss-of-coolant accident, main steam line break, fuel handling accident, or steam generator tube rupture), the following assumptions are made, in addition to postulating the initiating event.

a. The events are assumed not to result from a tornado, hurricane, flood, fire, loss of offsite power, or earthquake.

b. Any one of the following occurs:

   1. During the short term of an accident, a single failure of any active mechanical component. The short term is defined as less than 24 hours following an accident, or

   2. During the short term of an accident, a single failure of any active or passive electrical component, or

   3. A single failure of passive components associated with long-term cooling capability, assuming that a single active failure has not occurred during the short term. Long-term cooling applies to a time duration greater than 24 hours.

c. No reactor coolant system transient is assumed, preceding the postulated reactor coolant system piping rupture.

d. No operator action is assumed to be taken by plant operators to correct problems during the first 10 minutes following the accident. INSERT A

e. All offsite power is simultaneously lost and is restored within 7 days.

f. For a LOCA, for additional safety no credit is taken for the functioning of nonseismic Category I components.

INSERT A

Although not a design basis accident, operator action times of less than 10 minutes are assumed in the mitigation of an inadvertent ECCS actuation at power event. See Section 15.5.1.

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Westinghouse identified four control systems for generic consideration of nonsafety grade/safety grade interface interactions.

a. Steam generator power-operated relief valve control system - A piping failure in the vicinity of the steam generator relief valves could be assumed to cause the valves to stick open. The combination of the pipe failure, an assumed single failure, and the stuck open valve(s) may result in inadequate auxiliary feedwater flow.

   Westinghouse performed generic analyses for this type of event during development of the Emergency Response Guidelines (ERGs). ERG ECA 2.1, "Uncontrolled Depressurization of All Steam Generators" considers a worst-case multi-steam generator depressurization. The Westinghouse analysis, which bounds the relief valve opening event, concluded that a stabilized plant and safe cooldown can be achieved with a flow equivalent to one motor-driven auxiliary feedwater pump. Therefore, this scenario does not present a safety problem for the Callaway design.

b. Pressurizer power-operated relief valves control system - ~~A failure of secondary system piping inside the containment is assumed to cause pressurizer power operated relief valves (PORV) to open. The resultant secondary break coincident with PORV opening may have more severe consequences than those accidents previously analyzed.~~

   The Callaway and automatic actuation circuitry pressurizer PORV, ~~and~~ associated pressure transmitters, meet Class 1E requirements and are qualified to the postulated accident environments inside the containment. ~~The control circuitry in the non-safety process control cabinets, as shown in Figure 7.2-1, sheet 11, is located in the mild environment control room cabinet area. This control circuitry is isolated from the safety-related process protection cabinets by loop power supply cards. This control portion of the PORV circuitry is not subject to the environmental effects of the postulated secondary break.~~ Therefore, this scenario does not present a safety problem for the Callaway design.

c. Main feedwater control system - A small feedwater line break could affect normal feedwater flow control, causing low steam generator levels prior to protective actions for the break.

   The Callaway feedwater line break accident has been reanalyzed, assuming the control and protection grade system interaction. The analysis shows that this scenario can be accommodated without violating design conditions and acceptance criteria. A summary of the analysis may be found in Section 15.2.8. The summary includes an identification of the analysis assumptions that are different from those used in Reference 18.

d. Automatic rod control system - An intermediate size high energy line break is assumed to affect the rod control system, such that the initial conditions previously assumed for the break may not be valid.

CALLAWAY - SP

TABLE 5.4-11

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Psig
Hydrostatic test pressure	3,107
Design pressure	2,485
Safety valves (begin to open)	~~2,485~~ 2,460
High pressure reactor trip	2,385
High pressure alarm	2,310
Power-operated relief valves	2,335*
Pressurizer spray valves (full open)	2,310
Pressurizer spray valves (begin to open)	2,260
Proportional heaters (begin to operate)	2,250
Operating pressure	2,235
Proportional heater (full operation)	2,220
Backup heaters on	2,210
Low pressure alarm	2,210
Pressurizer power-operated relief and isolation valve interlock - auto closure.	2,185
Low pressure reactor trip	1,885

*At 2,335 psig, a pressure signal initiates actuation (opening) of these valves. Remote manual control is also provided.

TABLE 5.4-17

PRESSURIZER VALVES DESIGN PARAMETERS

Pressurizer Safety Valves	
Number	3
Minimum ~~Maximum~~ relieving capacity, at 2,485 psig, ASME rated flow, lb/hr	420,000
Set pressure, psig	2,460 ~~2,485~~
Design temperature, F	650
Fluid	Saturated steam
Transient condition, F	(Superheated steam) 680
Backpressure	
Normal, psig	3 to 5
Expected during discharge, psig	500
Environmental conditions	
Ambient temperature (F)	50 to 120
Relative humidity (%)	0 to 100
Pressurizer Power-Operated Relief Valves	
Number	2
Design pressure, psig	2,485
Design temperature, F	650
Relieving capacity at 2,335 psig, per valve, lb/hr	210,000
Fluid	Saturated steam
Transient condition, F	(Superheated steam) 680

adversely affecting the safety or operability of the plant;

b. The probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation; and

c. The actuated equipment can be routinely tested when the reactor is shut down.

The list of equipment that is not tested at full power so as not to damage equipment or upset plant operation is:

a. Manual actuation switches (RTS and ESFAS)

b. Main turbine trip system (actual trip)

c. Main steam isolation valves (actual full closure)

d. Main feedwater isolation valves (actual full closure)

e. Feedwater control valves (actual full closure)

f. Main feedwater pump trip solenoids

g. Reactor coolant pump seal water return valves (actual full closure)

h. ~~Seven~~ Eight selected slave relays

The justifications for not testing the above items at full power are discussed below.

a. Manual actuation switches for RTS and ESFAS

These would cause initiation of their protection system function at power, causing plant upset and/or reactor trip. It should be noted that the reactor trip function that is derived from the automatic safety injection signal is tested at power in the same manner as the other analog signals and as described in Section 7.2.2.2.3. The processing of these signals in the solid state protection system wherein their channel orientation converts to a logic train orientation is tested at power by the built-in semiautomatic test provisions of the solid state protection system. The reactor trip breakers are tested at power, as discussed in Section 7.2.2.2.3.

valve chatter. Valve chatter could damage this relief valve. Testing of these valves at power could cause equipment damage. Therefore, these valves will be tested during scheduled refueling outages. Thus, the guidelines of Regulatory Position D.4 of Regulatory Guide 1.22 are met.

h. Eight ~~Seven~~ selected slave relays

Slave relays K602, K620 (turbine trip circuitry only; main feedwater pump trip solenoid circuitry is excluded as discussed in f above), K622, K624, K630, K740, ~~and~~ K741, and K750 and their actuated equipment will be tested at least once per 18 months during refueling and during each cold shutdown exceeding 24 hours unless they have been tested within the previous 92 ~~90~~ days. Justification for the extended test interval is based on plant operational concerns and was presented in detail in References 3, 6 and 5.

7.1.2.6 Conformance to IEEE Standards

7.1.2.6.1 Conformance to IEEE Standard 379-1972

The principles described in IEEE Standard 379-1972 were used in the design of the solid state protection system. The system complies with the intent of this standard and the additional guidance of Regulatory Guide 1.53, although the formal analyses have not been documented exactly as outlined. Westinghouse has gone beyond the required analyses and has performed a fault tree analysis (Ref. 4).

The referenced report provides details of the analyses of the solid state protection system previously made to show conformance with the single failure criterion set forth in Section 4.2 of IEEE Standard 279-1971. The interpretation of the single failure criterion provided by IEEE Standard 379-1972 does not indicate substantial differences with the Westinghouse interpretation of the criterion, except in the methods used to confirm design reliability. The RTS and ESFAS are each redundant safety systems. The required periodic testing of these systems will disclose any failures or loss of redundancy which could have occurred in the interval between tests, thus ensuring the availability of these systems.

7.1.2.6.2 Conformance to IEEE Standard 338-1971

The periodic testing of the RTS and ESFAS conforms to the requirements of IEEE Standard 338-1971 with the following comments:

a. The surveillance requirements in the Callaway Technical Specifications for the solid state protection system ensure that the system functional operability is maintained comparable to the original design standards.

e. The test interval discussed in Section 5.2 of IEEE Standard 338-1971 is developed primarily on past operating experience and modified, if necessary, to ensure that system and subsystem protection is reliably provided.

7.1.3 REFERENCES

1. Marasco, F.W. and Siroky, R.M., "Westinghouse 7300 Series Process Control System Noise Tests," WCAP-8892-A, June, 1977.

2. Letter dated April 20, 1977, R.L. Tedesco (NRC) to C. Eicheldinger (Westinghouse).

3. Letter dated February 27, 1984, N.A. Petrick (SNUPPS) to Mr. Harold R. Denton (NRC), SLNRC 84-0038.

4. Gangloff, W.C. and Loftus, W.D., "An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients," WCAP-7706-L (Proprietary) and WCAP-7706 (Non-Proprietary), July, 1971.

5. Operating License Amendment XXX dated

7.2.2.3.3 Pressurizer Pressure

The pressurizer pressure protection channel signals are used for high and low pressure protection and as inputs to the overtemperature ΔT trip protection function. Isolated output signals from these channels are used for pressure control. These are used to control pressurizer spray and heaters. ~~and~~ Safety-related automatic actuation signals are also used to actuate the power-operated relief valves. Pressurizer pressure is sensed by fast response pressure transmitters.

A spurious high pressure signal from one channel can cause decreasing pressure by actuation of either spray or relief valves. Additional redundancy is provided in the low pressurizer pressure reactor trip and in the logic for safety injection to ensure low pressure protection.

Overpressure protection is based upon the positive surge of the reactor coolant produced as a result of turbine trip under full load, assuming that the core continues to produce full power. The self-actuated safety valves, with a nominal set pressure of 2460 psig, are sized on the basis of steam flow from the pressurizer to accommodate this surge at a ~~setpoint~~ pressure of 2,500 psia and an accumulation of 3 percent. No credit is taken for the relief capability provided by the power-operated relief valves during this surge.

In addition, operation of either of the power-operated relief valves can maintain pressure below the high pressure trip setpoint for most transients. The rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available to alert the operator of the need for appropriate action.

Redundancy is not compromised by having a shared tap (see Section 7.2.1.1.2) since the logic for this trip is two out of four. If the shared tap is plugged, the affected channels will remain static. If the impulse line bursts, the indicated pressure will drop to zero. In either case, the fault is easily detectable, and the protective function remains operable.

7.2.2.3.4 Pressurizer Water Level

Three pressurizer water level channels are used for reactor trip. Isolated signals from these channels are used for pressurizer water level control. A failure in the level control system could fill or empty the pressurizer at a slow rate (on the order of half an hour or more).

The high water level trip setpoint provides sufficient margin so that the undesirable condition of discharging liquid coolant through the safety valves is avoided. Even at full power conditions, which would produce the worst thermal expansion rates, a failure of the water level control would not lead to any liquid discharge through the safety valves. This is due

(2) One wide range RCS pressure signal derived from a channel in a Train A related protection set.

b. Pressure and Temperature Inputs to PCV456A

(1) Four wide range RCS temperature signals derived from channels in a Train B related protection set.

(2) One wide range RCS pressure signal derived from a channel in a Train B related protection set.

The wide range RCS temperatures in each protection set are auctioneered in an auctioneering device in each protection set to select the lowest reading.

mitigation system (COMS)

An alarm is actuated when the auctioneered low temperature from the RCS wide range temperature channels falls within the range of cold overpressure applicability, thereby alerting the operator to arm the RCS cold overpressure mitigation system. ~~which automatically opens the block valve when the block valve control switch is in the automatic position.~~

The lowest reading is selected and input to a function generator which calculates the reference pressure limit program, considering the plant's allowable pressure and temperature limits. Also available from the related protection set is the wide range RCS pressure signal. The reference pressure from the function generator is compared to the actual RCS pressure monitored by the wide range pressure channel. The error signal derived from the difference between the reference pressure and the actual measured pressure will first annunciate a main control board alarm whenever the actual measured pressure approaches, within a predetermined amount, the reference pressure. On a further increase in measured pressure, the error signal will generate an actuation signal.

~~Logic is also provided to close the block valve automatically if the relief valve fails or sticks in the open position following some plant transient when the RCS temperature is above the cold overpressurization setpoint, and the RCS pressure drops below the reset pressure for the relief valve.~~

The monitored generating station variables that generate the actuation signal for the redundant PORV are processed in a similar manner.

Upon receipt of the actuation signal, the actuation device will automatically cause the PORV to open. Upon sufficient RCS inventory letdown, the operating RCS pressure will decrease, clearing the actuation signal. Removal of this signal causes the PORV to close.

7.6.6.1 Analysis of Interlocks

Many criteria presented in IEEE Standards 279-1971 and 338-1971 do not apply to the interlocks for RCS pressure control during low temperature operation, because the interlocks do not perform a protective function but, rather, provide automatic pressure control at low temperatures as a back-up to the operator. However, although IEEE Standard 279-1971 criteria do not apply, some advantages of the dependability and benefits of an IEEE Standard 279-1971 design have accrued by including selected elements, as noted above, in the protection sets and by organizing the control of the two PORVs (either of which can accomplish the RCS pressure control function) into dual channels.

The design of the low temperature interlocks for RCS pressure control is such that pertinent features include:

a. No credible failure at the output of the protection set racks, after the output leaves the racks to interface with the interlocks, will prevent the associated protection system channel from performing its protective function because of the separation of Train B interlocks from Train A (see Figure 7.6-4).

b. Testing capability for elements of the interlocks within (not external to) the protection system is consistent with the testing principles and methods discussed in Section 7.2.2.2.3, item J. It should be noted that there is an annunciator which provides an alarm when the ~~block valve~~ is armed coincident with a closed position of the motor-operated (MOV) pressurizer relief block valve. This MOV is in the same fluid path as the PORV, with a separate MOV and alarm used with the second PORV. COMS

c. A loss of offsite power will not defeat the provisions for an electrical power source for the interlocks because these provisions are through onsite power, which is described in Section 8.3.

7.6.7 ISOLATION OF ESSENTIAL SERVICE WATER (ESW) TO THE AIR COMPRESSORS

7.6.7.1 Description

As stated in Section 9.2.1.2.2.1, ESW flow to the nonsafety-related air compressors and associated aftercoolers is maintained following a DBA. Instrumentation and controls are provided to automatically isolate each train of the ESW to the air compressors on high flow. ESW to the air compressors can also be isolated by remote manual means.

the potential for impairing reactor vessel integrity when operating at or near the vessel ductility limits.

b. Capability for RCS depressurization following Condition II, III, and IV events (e.g., see Sections 15.5.1 and 15.6.3).

c. ~~Interlock that, with the pressurizer PORVs and PORV block valves in auto control, closes the PORV block valves and prevents signals from the pressurizer pressure control system from opening the PORVs when pressurizer pressure is low.~~

7.6.10.2 <u>Description of Pressurizer Pressure Relief System Interlocks</u>

Interlocks for the PPR system control the opening and closing of the pressurizer PORVs. ~~and the PORV block valves.~~ These interlocks provide the following functions:

a. Pressurizer pressure control via Class IE automatic actuation circuitry (refer to Section 7.7.1.5 for a description).

b. RCS pressure control during low temperature operation (refer to Sections 5.2.2 and 7.6.6 for a description).

c. RCS pressure control to achieve and maintain a cold shutdown and to heatup, using equipment that is required for safety (refer to Appendix 5.4A for a description).

The interlock functions that provide pressurizer pressure control are derived from process parameters as shown on Figure 7.2-1, Sheet 11 and the interlock logic functions as well as process parameter inputs required for low temperature operation, as shown on Figure 7.6-4. The functions shown on Figure 7.6-4 include those needed for the PORV block valves as well as the pressurizer PORVs to meet both interlock logic and manual and automatic operation requirements where manual operation is at the main control board.

7.6.11 SWITCHOVER OF CHARGING PUMP SUCTION TO RWST ON LOW-LOW VCT LEVEL

7.6.11.1 <u>Description</u>

The suction of the charging pumps is normally supplied by a line containing two normally open motor-operated valves which connects to the bottom of the volume control tank (VCT). These VCT outlet isolation valves are designated as LCV-112B, which is assigned to the A train, and LCV-112C, which is assigned to the B train.

7.7.1.5 Pressurizer Pressure Control

The reactor coolant system pressure is controlled by using either the heaters (in the water region) or the spray (in the steam region) of the pressurizer plus steam relief for large transients.

The electrical immersion heaters are located near the bottom of the pressurizer. A portion of the heater group is proportionally controlled to correct small pressure variations. These variations are caused by heat losses, including heat losses due to a small continuous spray. The remaining (back-up) heaters are turned on when the pressurizer pressure controlled signal demands approximately 100-percent proportional heater power.

The spray nozzle is located on the top of the pressurizer. Spray is initiated when the pressure controller spray demand signal is above a given setpoint. The spray rate increases proportionally with increasing spray demand signal until it reaches a maximum value.

Steam condensed by the spray reduces the pressurizer pressure. A small continuous spray is normally maintained to reduce thermal stresses and thermal shock and to help maintain uniform water chemistry and temperature in the pressurizer.

Power relief valves limit system pressure for large positive pressure transients. In the event of a large load reduction, not exceeding the design plant load rejection capability, the pressurizer power-operated relief valves might be actuated for the most adverse conditions, e.g., the most negative Doppler coefficient and the maximum incremental rod worth. The relief capacity of the power-operated relief valves is sized large enough to limit the system pressure to prevent actuation of high pressure reactor trip for the above condition. The automatic actuation circuitry for the PORVs has been upgraded to Class 1E. A block diagram of the pressurizer pressure control system is shown on Figure 7.7-4.

7.7.1.6 Pressurizer Water Level Control

The pressurizer operates by maintaining a steam cushion over the reactor coolant. As the density of the reactor coolant varies with temperature, the steam water interface is adjusted to compensate for cooling density variations with relatively small pressure disturbances.

values. Additionally, protection system channels are calibrated and instrument response times determined periodically in accordance with the Technical Specifications.

15.0.7 INSTRUMENTATION DRIFT AND CALORIMETRIC ERRORS - POWER RANGE NEUTRON FLUX

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 15.0-5.

The calorimetric error is the error assumed in the determination of core thermal power, as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

The secondary power is obtained from the measurement of steam or feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. High accuracy instrumentation is provided for use during these measurements. Accuracy tolerances meet or exceed requirements established by the safety analysis.

15.0.8 PLANT SYSTEMS AND COMPONENTS AVAILABLE FOR MITIGATION OF ACCIDENT EFFECTS

The plant is designed to afford protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. The Operating Quality Assurance Manual discusses the quality assurance program which has been implemented to assure that the NSSS will satisfactorily perform its assigned safety functions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-6 will be available for mitigation of the events discussed in Chapter 15.0. In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI- N18.2-1973 is utilized. The design of safety-related systems (including protection systems) is consistent with IEEE Standard 379-1972 and Regulatory Guide 1.53, in the application of the single failure criterion.

In the analysis of the Chapter 15.0 events, control system action is considered only if that action results in more severe accident results. No credit is taken for control system operation if that operation mitigates the results of an accident. For some accidents, the analysis is performed both with and without control system operation to determine the worst case. The pressurizer heaters are not assumed to be energized during any of the Chapter 15.0 events, except as discussed in Section 15.5.1.

taken by the operator would be no different than normal operating procedures. The exact actions taken, and the time at which these actions would occur, will depend on what systems are available (e.g., turbine bypass system, main feedwater system, etc.) and the plans for further plant operation. As a minimum, to maintain the hot stabilized condition, decay heat must be removed via the steam generators. The main feedwater system and the steam dump or atmospheric relief system could be used for this purpose. Alternatively, the auxiliary feedwater system and the steam generator safety valves may be used, both of which are safety grade systems. Although the auxiliary feedwater system may be started manually, it will be automatically actuated, if needed, by one of the signals shown on Figure 7.3-1 (sheet 2), such as low-low steam generator water level. Also, if the hot standby condition is maintained for an extended period of time (greater than approximately 18 hours), operator action may be required to add boric acid via the CVCS to compensate for xenon decay and maintain shutdown margin.

Where a stabilized condition is reached automatically following a reactor trip and only actions typical of normal operation are required, this has been stated in the text of the Chapter 15.0 events. For several events involving breaks in the reactor coolant system or secondary system piping, additional requirements for operator action are identified.

→ INSERT B

Following the postulated MSLB, a steamline isolation signal will be generated almost immediately, causing the main steam isolation valves to close within a few seconds. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the break is upstream of the isolation valves, or if one valve fails to close, the break will be isolated to three steam generators while the affected steam generator will continue to blow down. Only the case in which one steam generator continues to blow down is discussed here, since the break followed by isolation of all steam generators will terminate the transient.

Steam pressure from the steam generators is relieved by the turbine bypass system, secondary system atmospheric safety valves, or secondary system atmospheric relief valves (also referred to as the atmospheric steam dump valves). The operator is instructed to terminate auxiliary feedwater flow to the affected steam generator, as soon as he determines which steam generator is affected. As soon as an indicated water level returns to the pressurizer and pressure is no longer decreasing, the operator is instructed to terminate the charging pump flow to limit system repressurization.

For long-term cooling following a steamline break, the operator is instructed to use the intact steam generators for the purpose of removing decay heat and plant stored energy. This is done by feeding the steam generators with auxiliary feedwater to maintain an indicated water level in the steam generator narrow-range span.

INSERT B

For the inadvertent ECCS actuation at power event, timely operator actions are completed from the main control room to terminate NCP flow and to open PORV block valve(s) and assure the availability of the PORV(s) for automatic pressure relief to allow automatic actuation of at least one PORV on demand. These actions will preclude water relief through the pressurizer safety valves. The required operator action times are discussed in Section 15.5.1 and Table 15.5-1.

TABLE 15.0-6 (Sheet 4)

	Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment/Alarms	ESF Equipment
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	Source range high flux (1-Mode 2), power range high flux, overtemperature ΔT (1-Mode 1)	---	BDMS-activated CCP suction valve swapover from VCT to RWST in Modes 3-5, at least one RCP in operation in Modes 3-5, (see Section 15.4.6.1 for complete list of status and annunciator indications)	---
	Spectrum of rod cluster control assembly (RCCA) ejection accidents	Power range high flux (high and low setpoints) (1), high positive flux rate, manual	SIS on low pressurizer pressure	---	Safety injection system
15.5	Increase in reactor coolant inventory				
	Inadvertent operation of the ECCS during power operation	Low pressurizer pressure (1-DNB Case), manual, safety injection trip (1, Pressurizer Filling Case)	---	---	Safety injection system (transient initiator)
	CVCS malfunction that increases reactor coolant inventory (3)	High pressure water level (4)	---	High pressurizer pressure and water level annunciators, pressurizer PORVs	---
15.6	Decrease in reactor coolant inventory				
	Inadvertent opening of a pressurizer safety or relief valve	Low pressurizer pressure, overtemperature ΔT (1), manual	---	---	---
	Steam generator tube failure	Low pressurizer pressure (1), SIS, manual, OTΔT (4)	SIS on low pressurizer pressure (1), AFAS and FWIS on SIS (1)	Essential service water system, component cooling water system, steam generator safety and/or relief valves, steam line isolation valves, feedwater isolation valves	Emergency core cooling system, auxiliary feedwater system, emergency power system

TABLE 15.0-6 (Sheet 5)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment/Alarms	ESF Equipment
Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure (1), SIS, OTΔT, manual	SIS on Hi-1 containment pressure or low pressurizer pressure (1), AFAS and FWIS on SIS (modeled only in SBLOCA analysis) (1)	Essential service water system, component cooling water system, steam generator safety and/or relief valves, feedwater isolation valves (modeled for SBLOCA only)	Emergency core cooling system, TD auxiliary feedwater pump (modeled for SBLOCA only), containment heat removal system, emergency power system

Notes:

(1) Trip function credited in the safety analysis. Some analyses credit different trip functions depending on different analysis cases presented.

(2) Turbine trip on high-high steam generator water level not directly credited in the Section 15.1.2 analysis.

(3) No automatic protection action required.

(4) Trip function specifically excluded from potentially available functions to drive analysis in a given direction.

(5) Reactor trip on safety injection signal is modeled at the start of the transient to minimize the time to fill the pressurizer.

offsite power). It is apparent from the initial portion of the transient (~150 seconds), that the case without offsite power results in higher temperatures in the hot leg. For longer times, however, the case with offsite power results in a more severe rise in temperature until the coolant pumps are turned off and the auxiliary feedwater system is realigned. AFWS realignment is not assumed in these analyses. The pressurizer fills more rapidly for the case with power due to the increased coolant expansion resulting from the pump heat addition. As previously stated, however, the core remains covered with water for both cases.

15.2.8.3 Analysis of Effects and Consequences

Method of Analysis (No SI Cases)

As a sensitivity on the above licensing basis cases, the effect of having no safety injection in the above feedwater pipe rupture cases (with and without offsite power) was analyzed. In addition, a part-power break case (29% rated thermal power with offsite power) was analyzed to evaluate the effects on Reference 6.

These cases were analyzed with the LOFTRAN code (Reference 2) with the same analysis assumptions as used in Section 15.2.8.2 except:

a. No safety injection system actuation occurs.

b. Steam and water are relieved from the pressurizer power-operated relief valves and safety valves.

With the assumption of no safety injection it is conservative to assume that the ~~non-1E~~ pressurizer PORVs are operational, along with the safety valves, since a lower RCS pressure would correspond to lower hot leg saturation temperatures, the second acceptance criterion discussed in Section 15.2.8.1. Lower RCS pressure would also provide a faster heatup of the primary side since the coolant density is lower. Further, water relief through the PORVs results in a lower water inventory in the primary side and faster heatup of the RCS. The licensing basis cases discussed in Section 15.2.8.2 assume relief only from the pressurizer safety valves since the higher predicted RCS pressures lead to lower SI flows during the transient; there is no need to force the transient in that direction here since no SI is assumed.

Results (No SI Cases)

Calculated plant parameters are shown in Figures 15.2-25 through 15.2-46. Results for the case with offsite power available are presented in Figures 15.2-25 through 15.2-35. Results for the case where offsite power is lost are presented in Figures 15.2-36 through 15.2-46. The calculated sequences of events for both cases analyzed are listed in Table 15.2-1.

15.5 INCREASE IN REACTOR COOLANT INVENTORY

Discussion and analysis of the following events are presented in this section:

a. Inadvertent operation of the emergency core cooling system during power operation.

b. Chemical and volume control system malfunction that increases reactor coolant inventory.

c. A number of BWR transients. (Not applicable to Callaway.)

These events, considered to be ANS Condition II, cause an increase in reactor coolant inventory. Section 15.0.1 contains a discussion of ANS classifications.

15.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

15.5.1.1 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 actuation signal. A spurious signal may originate from any of the safety injection actuation channels, as described in Section 7.3.

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 valves isolating the boron injection header (BIH) from the centrifugal charging pumps and the valves isolating the BIH from the injection header then automatically open. The centrifugal charging pumps then inject boric acid solution into the cold leg of each loop. The safety injection pumps also start automatically but provide no flow when the reactor coolant system (RCS) is at normal pressure. The passive, accumulator safety injection system and the RHR system also provide no flow at normal RCS pressure.

A safety injection signal (SIS) normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates ECCS will also produce a reactor trip. If a reactor trip is generated by the spurious SIS, the operator should determine if the spurious signal was transient or steady state in nature. The operator must also determine if the SIS should be blocked. For a spurious occurrence, the operator would, per procedure, terminate ECCS operation and maintain the plant in the hot shutdown condition after completing the actions described in Section 15.5.1.2 item J. If the ECCS actuation instrumentation must be repaired, subsequent plant operation would be in accordance with the Technical Specifications.

condition of the plant operating with all the PORVs blocked, actions to ~~terminate the ECCS flow to avert a water-solid condition must be taken.~~ open at least one PORV block valve and ensure the PORV is available for automatic pressure relief must be taken within 9 minutes after the transient begins. The Inadvertent ECCS Actuation at Power Event is analyzed to determine ~~both~~ the minimum DNBR value and ~~maximum pressurizer water volume (or minimum time to~~ to demonstrate the adequacy of plant procedures to prevent water relief through the safety valves under a pressurizer water-solid condition~~)~~. 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.

The analysis assumptions are as follows (see also Section 15.0.3 and Table 15.0-2):

A. Initial Operating Conditions

The DNB case is analyzed with the Improved Thermal Design Procedure as described in Reference 2. 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 2.

For the pressurizer filling case, a non-ITDP analysis, initial conditions with maximum uncertainties on power (+2%), vessel average temperature (-7.5°F), and pressurizer pressure (-30 psia) are assumed. The lower initial temperature results in a higher RCS coolant mass which causes a more severe pressurizer water volume transient.

B. Moderator and Doppler Coefficients of Reactivity

The minimum feedback case (DNB) assumes a positive (+5 pcm/°F) moderator temperature coefficient and a low absolute value Doppler power coefficient. The maximum feedback case (pressurizer filling) assumes a large (absolute value) negative moderator temperature coefficient (0.43Δk/k/gm/cc is the corresponding moderator density coefficient) and a most-negative Doppler power coefficient.

C. Reactor Control

For the DNB case (without direct reactor trip on SI) the reactor is assumed to be in a manual rod control. In the case of the pressurizer filling scenario, the reactor is assumed to trip at the time of the SI signal. Thus, the reactor control mode is of no consequence.

INSERT C

~~D. Pressurizer Pressure Control~~

~~Pressurizer heaters are assumed to be inoperable. This assumption yields a higher rate of pressure decrease for the~~

INSERT C

D. Pressurizer Pressure Control

Pressurizer heaters are assumed to be inoperable for the DNB case. This assumption yields a higher rate of pressure decrease. Pressurizer sprays and PORVs are assumed to be available for this case in order to minimize RCS pressure.

Pressurizer sprays are assumed to be operable for the pressurizer filling case. Since the PORVs are assumed as an automatic pressure control function for this case, operator action to assure their availability for automatic pressure relief is assumed. Timely operation of the PORVs results in the pressurizer pressure not reaching the pressurizer safety valve set pressure such that the potential for water relief through the safety valves is precluded. Proportional heaters are assumed to remain operable throughout the transient (backup heaters are load shed upon receipt of the SI signal).

It should be noted that the analysis performed places no new requirements on the PORV opening setpoint. It simply assumes that at least one PORV will open prior to pressurizer pressure reaching the lowest PSV setpoint (accounting for negative uncertainty). As such, the analysis assumes the nominal PORV setpoint of 2335 psig. All applicable delays associated with the PORVs and block valves are accounted for in the assumed operator action time.

~~DNB case. Pressurizer spray is assumed available for each case in order to minimize RCS pressure. PORVs are also assumed operable in the DNB case.~~

~~PORVs are not assumed as an automatic pressure control function for the pressurizer filling case. Operation of the PORVs results in the pressurizer pressure decreasing below the safety valve set pressure prior to reaching a water-solid condition (i.e., the event criterion to preclude a water-solid pressurizer condition concurrent with the pressurizer at or above the safety valve set pressure would be met).~~

E. Boron Injection

At the initiation of the event, two ECCS charging pumps inject borated water into the cold leg of each loop. In addition, flow is also modeled to conservatively account for possible operation of the normal charging pump. This latter assumption is conservative since the normal charging pump receives a non-Class 1E trip signal after an SIS which is backed up by explicit operation action as directed by EOP E-0. The analysis assumes zero injection line purge volume for calculational simplicity; thus the boration transient begins immediately in the analysis.

F. Turbine Load

For the DNB 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 full 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.

G. Reactor Trip

Reactor trip is initiated by a low pressurizer pressure signal at 1860 psia for the DNB case. The pressurizer filling case assumes an immediate reactor trip on the initiating SI signal.

H. Decay Heat

Core residual heat generation is based on the 1979 version of ANS 5.1 (Reference 3). ANSI/ANS-5.1-1979 is a conservative representation of the decay energy release rates. Long-term operation at the initial power level preceding the trip is assumed.

→ INSERT D

~~I. Pressurizer Safety Valves~~

~~The safety valves open at a pressure of 2475 psia which corresponds to a tolerance of 1% relative to the set pressure of 2500 psia. The valves are assumed to close at a pressure of 2375 psia which corresponds to a blowdown of 5% below the set pressure of 2500 psia.~~

INSERT D

I. Pressurizer Safety Valves

The pressurizer safety valve opening pressure is assumed to be 2425 psia, which accounts for a -2% setpoint tolerance. Although the safety valves are not actuated during this transient, the assumed safety valve opening setpoint serves as a limit to demonstrate the acceptability of the assumptions made in item J below.

J. Operator Action Times

It is assumed that operator action is taken from the main control room such that flow from the normal charging pump is terminated within 6 minutes after the start of the transient.

The pressurizer safety valves must not be exposed to subcooled liquid discharge as a result of reaching a water solid pressurizer condition. Consequently, PORV availability must be assured by operator actions from the main control room to assure at least one PORV will actuate on demand. The PORVs would be expected to be available unless they were blocked due to excessive seat leakage. Therefore, the assumed operator actions associated with assuring PORV availability consist of opening a block valve(s) and assuring the PORV handswitch(es) are in the automatic operation position from the main control room to allow the PORV(s) to actuate on demand. The analysis assumes that appropriate operator action is taken to assure that at least one PORV is available for pressure relief within 9 minutes after the start of the transient. This time includes all process and instrumentation delays.

Results

The transient responses for the DNB and pressurizer filling cases are shown in Figures 15.5-1 through 15.5-3. Table 15.5-1 shows the calculated sequence of events.

DNB Case:

Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valve is wide open. The mismatch between load and nuclear power causes Tavg, pressurizer water level, and pressurizer pressure to drop. The reactor trips and control rods start moving into the core when the pressurizer pressure reaches the low pressurizer pressure trip setpoint. The DNBR remains above its initial value throughout the transient.

Pressurizer Filling Case:

RCS

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 heat generation, and ECCS injected flow causes the pressure and level transients to rapidly turn around. Pressurizer water level then increases throughout the transient. ~~The analysis assumes that the source of SI flow is isolated via operator action within 10 minutes. Results of the analysis demonstrate that the pressurizer does not fill and water relief through the pressurizer safety valves is precluded.~~

INSERT E

15.5.1.3 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 excessive cooldown, which expedites recovery from the incident.

INSERT F

~~With respect to pressurizer filling, the pressurizer will not reach a water-solid condition provided the source of SI flow is isolated within 10 minutes of the start of the transient. Water relief through the pressurizer safety valves is therefore precluded.~~

15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

15.5.2.1 Identification of Causes and Accident Description

Increases in reactor coolant inventory caused by 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 constant boron concentration. The transients analyzed in this

INSERT E

At six (6) minutes into the transient, the analysis assumes that normal charging pump flow is terminated via operator action from the main control room. The pressurizer becomes water solid approximately 8 minutes into the transient. At nine (9) minutes into the transient, it is assumed that appropriate operator actions have been taken from the main control room to assure that at least one PORV is available for automatic pressure relief. These operator action times include all process and instrumentation delays. At that time, the PORV(s) actuate and the RCS rapidly depressurizes. At no time is the safety valve setpoint challenged. Therefore, the analysis demonstrates that water relief through the pressurizer safety valves is precluded.

INSERT F

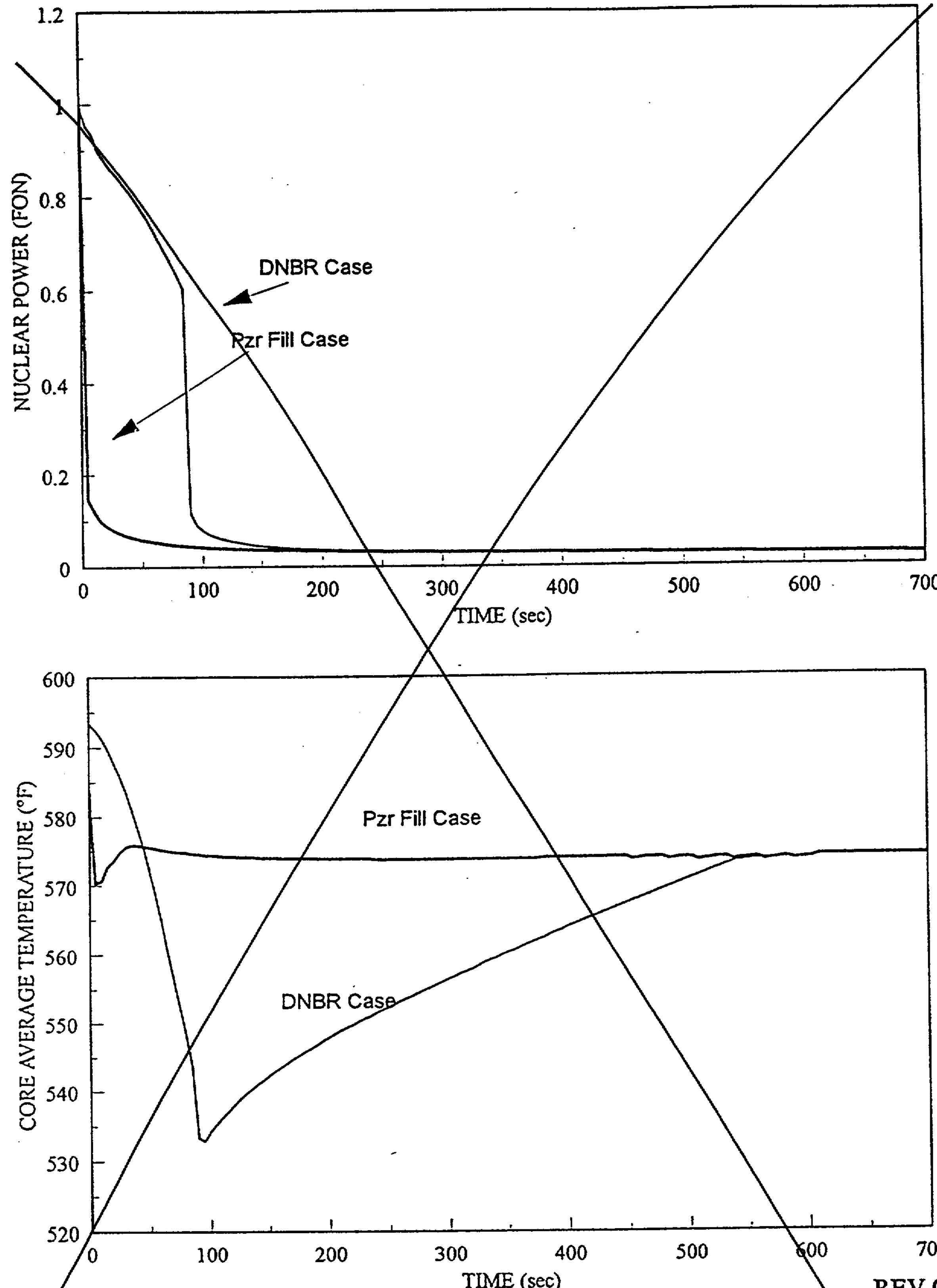
With respect to pressurizer filling, although the pressurizer may reach a water-solid condition, operator actions taken from the main control room to isolate the normal charging pump and assure that at least one PORV is available for automatic pressure relief will prevent the safety valves from actuating with the pressurizer in a water-solid condition. Water relief through the pressurizer safety valves is, therefore, precluded.

TABLE 15.5-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN AN INCREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)
Inadvertent Operation of ECCS During Power Operation		
DNBR Case	Charging pumps begin injecting borated water	0.0
	Low pressurizer pressure reactor trip setpoint reached	~~82.8~~ 85.0
	Rods begin to drop	~~84.8~~ 87.0
	Minimum DNBR occurs	(a)
Pressurizer Filling Case	Charging pumps begin injecting borated water; rods begin to drop	0.0
	~~Source of SI flow is isolated~~	~~600.0~~
	~~Peak pressurizer water volume~~	~~609.0~~
	NCP flow terminated	360
	Pressurizer filling occurs	478
	Automatic pressure relief via at least one PORV occurs	540

(a) DNBR does not decrease below its initial value



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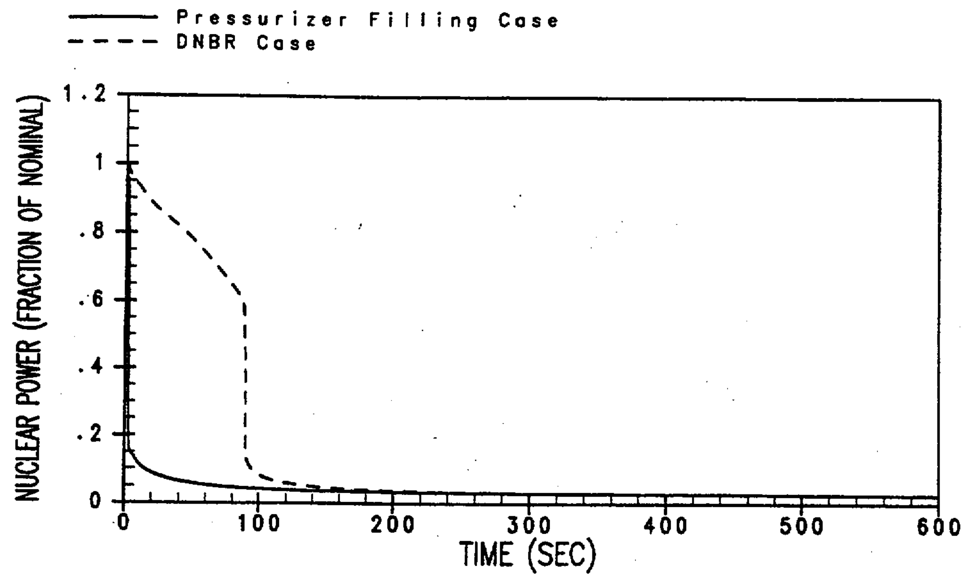
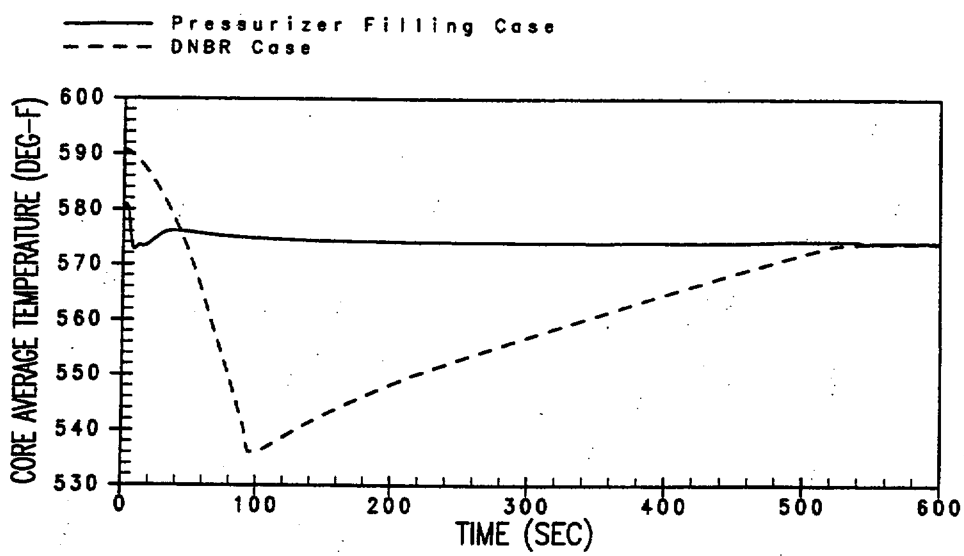
CALLAWAY PLANT
FIGURE 15.5-1 INADVERTENT OPERATION OF ECCS DURING POWER OPERATION

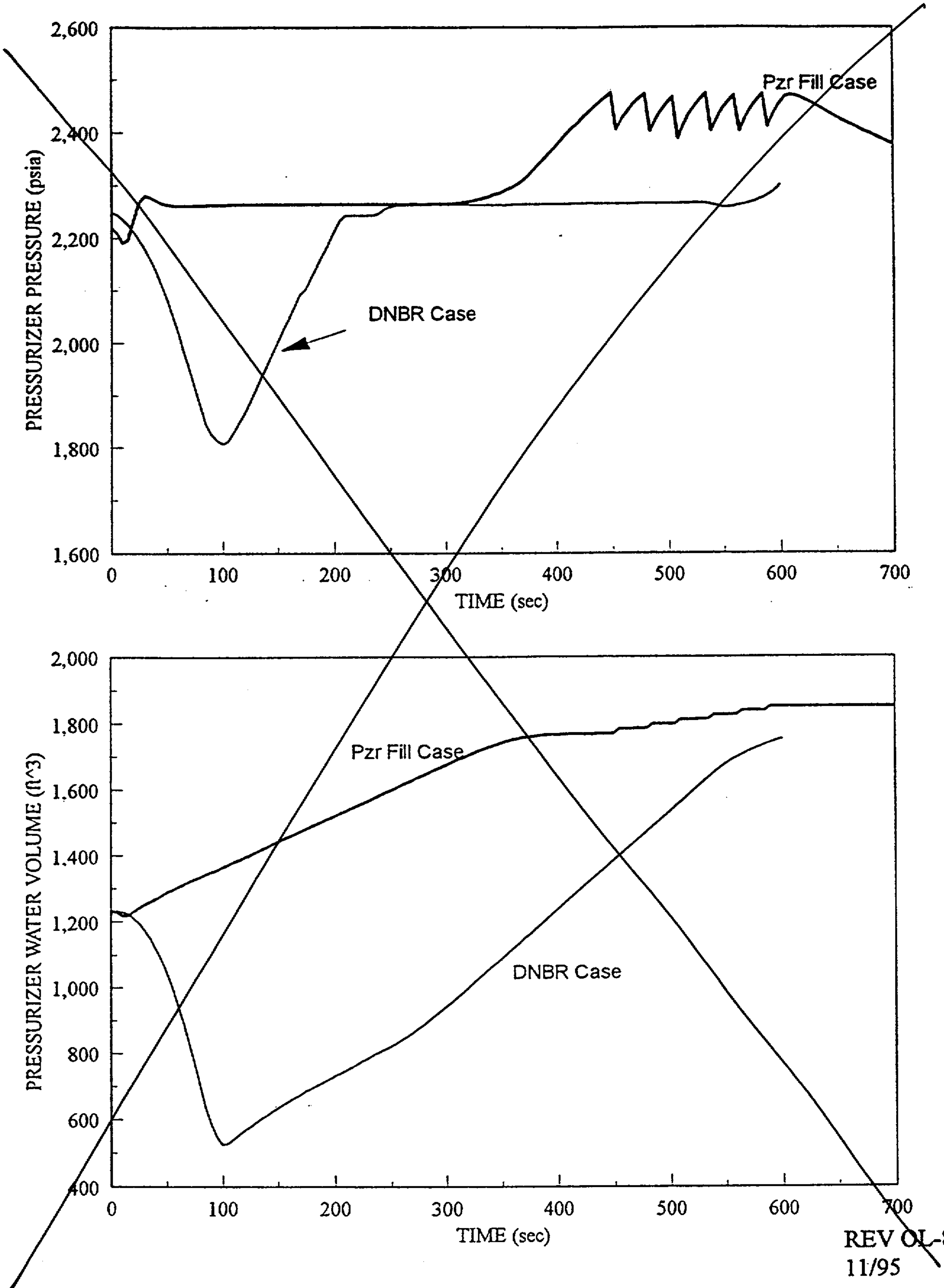




CALLAWAY PLANT

Inadvertent Operation of ECCS During Power Operation

Figure 15.5-1



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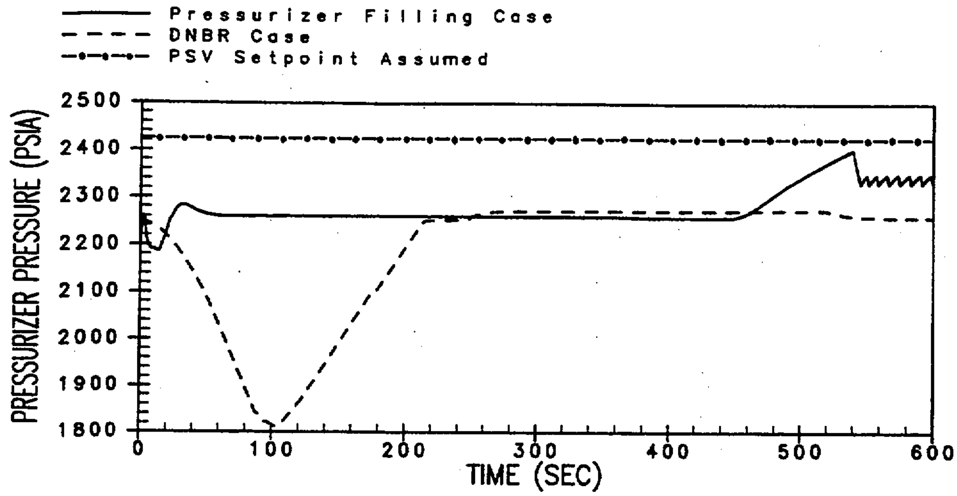
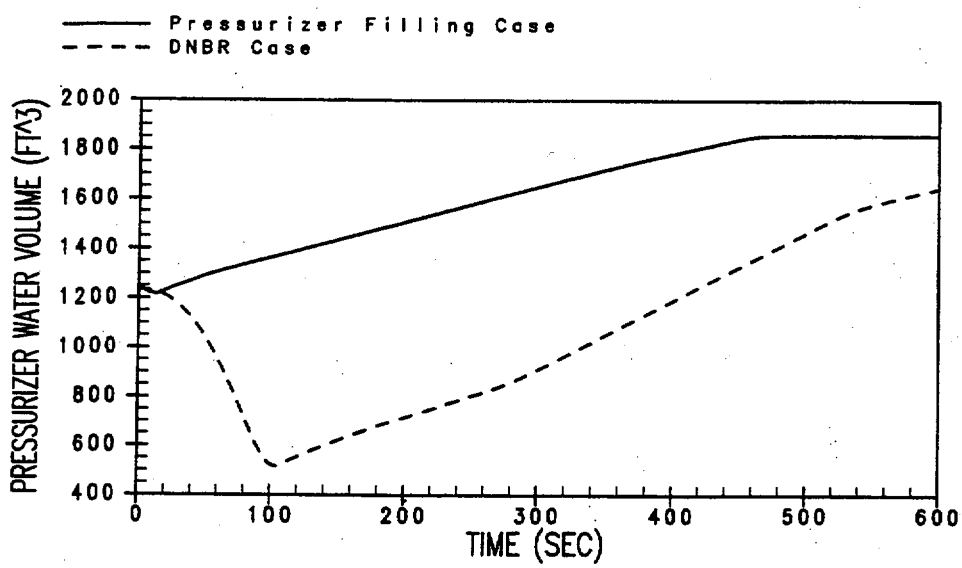
CALLAWAY PLANT

FIGURE 15.5-2
INADVERTENT OPERATION OF ECCS
DURING POWER OPERATION

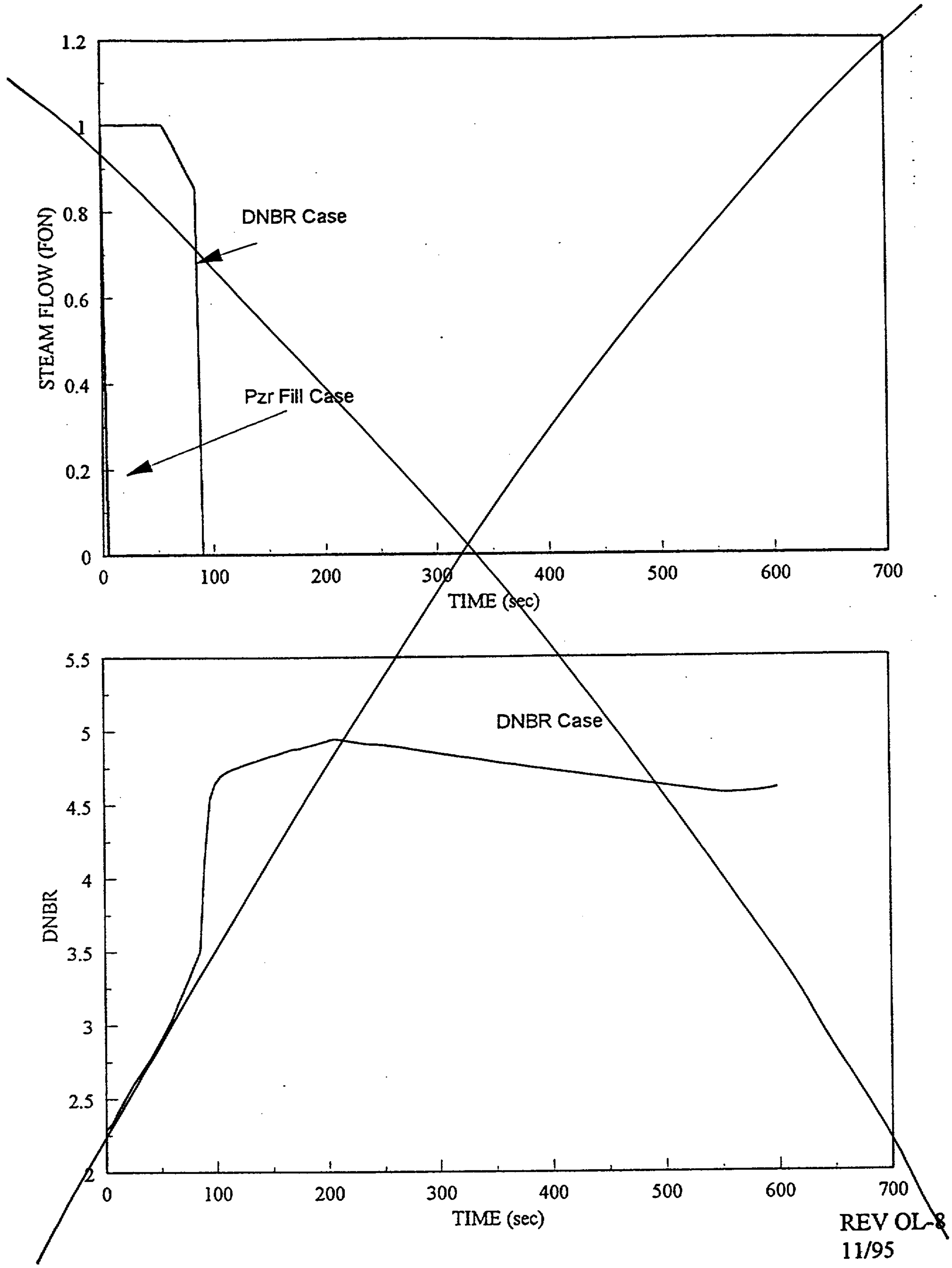




CALLAWAY PLANT

Inadvertent Operation of ECCS
During Power Operation

Figure 15.5-2



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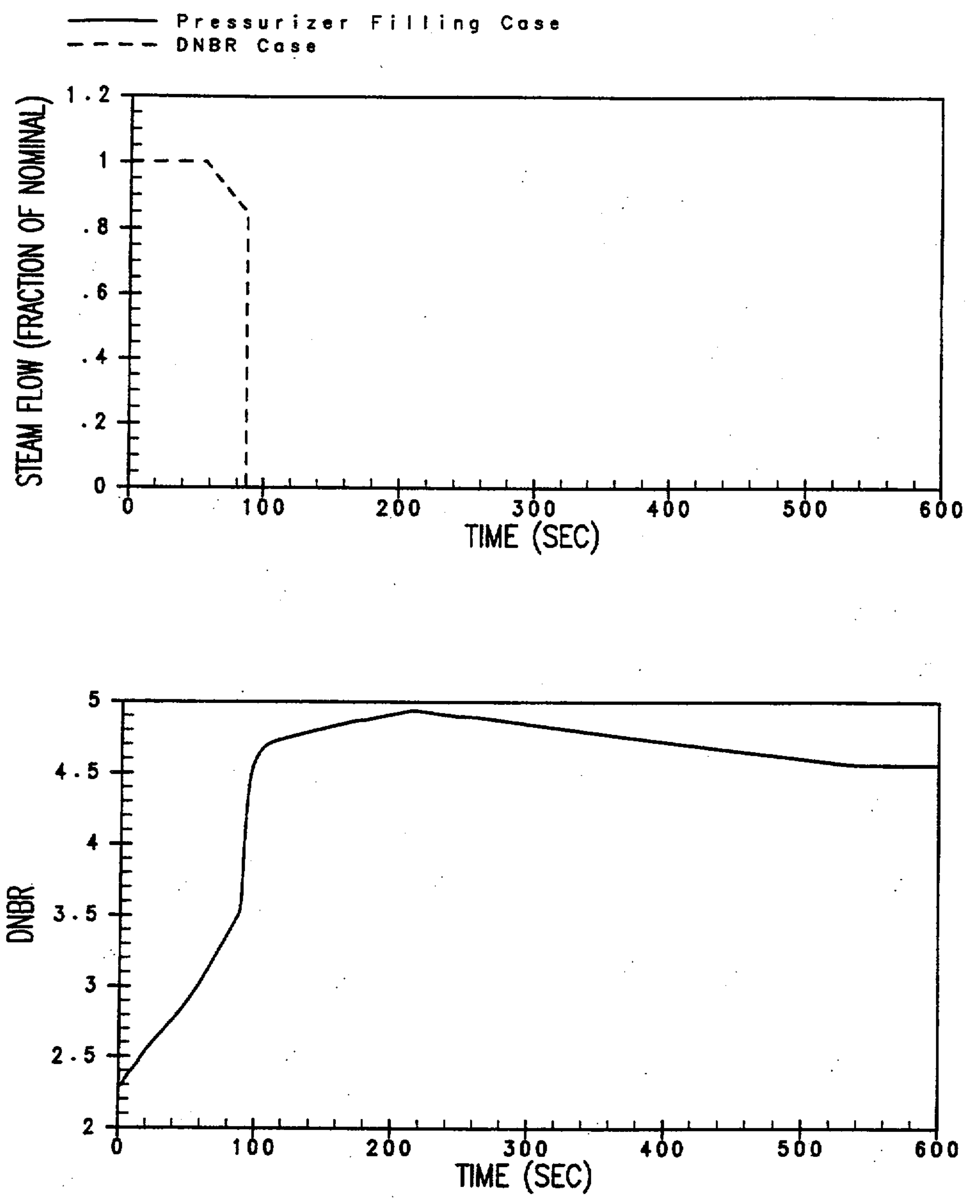
CALLAWAY PLANT

FIGURE 15.5-3
INADVERTENT OPERATION OF ECCS
DURING POWER OPERATION



CALLAWAY PLANT

Inadvertent Operation of ECCS
During Power Operation

Figure 15.5-3

FSAR C/N # 98-041

16.4 ~~(3/4.4)~~ REACTOR COOLANT SYSTEM

16.4.1 ~~(3/4.4.2)~~ SAFETY VALVES

16.4.1.1 ~~(3.4.2.1)~~ SHUTDOWN LIMITING CONDITION FOR OPERATION

≥ ~~2460~~ 2411 psig and ≤ ~~2510~~ 2509 psig GGG 3/29/99

A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of ~~2485 psig ±1%~~ (The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.)

APPLICABILITY: MODES 4 and 5. with any RCS cold leg temperature ≤ 275 degrees F

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

16.4.1.1.1 ~~(4.4.2.1)~~ SURVEILLANCE REQUIREMENTS

No additional requirements other than those required by ~~Section 16.0.2.5.~~ the Inservice Testing Program. The lift setting shall be within ±1% of 2460 psig following testing.

16.4.1.1.2 BASES

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Cold Overpressure ~~Protection~~ Mitigation System provides a diverse means of protection against RCS overpressurization at low temperatures.

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JMC 1-13-00

FSAR C/N # 00-013
CN #98-041

Insert 16.4.7

~~16.4.7 PRESSURIZER PORVs~~

~~16.4.7.1 LIMITING CONDITION FOR OPERATION~~

~~The PORV automatic actuation circuitry shall be OPERABLE for each Pressurizer PORV.~~

~~APPLICABILITY: MODES 1, 2, and 3.~~

~~ACTION:~~

~~With the PORV automatic actuation circuitry inoperable for one PORV, restore PORV automatic actuation circuitry to OPERABLE status within 72 hours; otherwise, enter Section 16.0.1.3.~~

~~16.4.7.1.1 SURVEILLANCE REQUIREMENTS~~

~~Perform a CHANNEL CALIBRATION of each PORV's automatic actuation instrumentation at least once per 18 months.~~

~~16.4.7.1.2 BASES~~

~~The pressurizer PORV's are equipped with automatic actuation circuitry and manual control capability. The automatic actuation of the PORVs is not credited in MODES 1, 2, or 3. The portions of the PORV actuation circuitry required for COMS are calibrated in accordance with Technical Specification SR 3.4.12.9. Although not required for MODES 1, 2, or 3, it is desirable to have the automatic pressure relief capability of the PORVs available during these MODES. The 72 hour AOT for one PORV inoperable is consistent with Technical Specification 3.4.11 Action B.3. Therefore, this specification assures that the automatic actuation circuitry is OPERABLE by performing a CHANNEL CALIBRATION at the stated frequency. The 18-month surveillance frequency is adequate based on operating experience.~~

JMC 3-20-00

~~The provisions of Section 16.0.1.4 are not applicable.~~

DELETED

2. The reactor vessel head vent system incorporates a 3/8-inch orifice to limit the maximum reactor coolant flow rate to a value less than that which defines a LOCA (see Figure 18.2-1).

The design provides for a motor-operated isolation valve in series with each pressurizer PORV. These PORV isolation valves are ~~may be either~~ remotely actuated from the control room~~, or automatically actuated based on an RCS pressure setpoint. The setpoint is selected based on providing isolation prior to actuation of the safety injection system.~~ Control room indication is provided for the pressurizer PORVs and PORV isolation valves and for the reactor vessel head vent valves. Each vent is remotely operable from the control room. An individual handswitch is provided for each valve.

The design of the RCS venting systems minimizes the probability of an inadvertent opening and consequence of such an opening.

1. The pressurizer vent system:

   The pressurizer PORVs are normally closed, Class IE solenoid valves that energize to open. Thus, loss of power will not actuate these valves. The PORV isolation valves are normally open, motor-operated valves. ~~As discussed above,~~ Assuming an inadvertent opening of the PORV or its failure to close, ~~a protection grade Class IE signal is provided to automatically~~ operator action is taken to close the associated block valve.

2. The reactor vessel head vent system:

   Each of the redundant vent paths off of the reactor vessel head contains two in-series, normally closed, same safety train, Class IE, environmentally qualified solenoid valves. The two normally closed valves in series limit any postulated events which could result in an inadvertent opening of the vent.

The pressurizer will vent to the pressurizer relief tank. The reactor vessel head vent system valves are located on the CRDM seismic support platform above the reactor vessel. The discharge from these valves will be directed to the open area of the containment above the refueling pool. This area precludes the potential for forming stagnant pockets of vented gases. Mixing and cooling of the vented gases will be accomplished using permanent plant systems.

The Westinghouse Owners Group (WOG) has developed a generic reactor vessel head vent guideline. The SNUPPS utilities will consider the generic guidance developed by the WOG in the development of procedures for use of the head vent system.

18.2.5.3 Discussion

The PORVs in the Callaway design are relied on to function to alleviate overpressurization that possibly could occur during startup of the reactor, during cold shutdown conditions, and they may be relied on to function during shut down of the reactor, assuming only safety-grade equipment is functioning. (These functions are described in Sections 5.2 and 5.4(A).) The PORVs are ~~not~~ required to function to mitigate the consequences of ~~any design basis accident.~~ the inadvertent ECCS actuation at power event.

The PORVs are also designed to limit high pressure during normal operation. The description of this control function is presented in Sections 5.2 and 7.6. As discussed below, operability of the PORVs will be demonstrated by prototypical testing and appropriate analyses.

The safety valves for the Callaway design are relied on to limit primary system pressure following anticipated operational transients. The design basis for the safety valves is presented in Section 5.2. The valves are required by ASME Boiler and Pressure Vessel Code to mitigate excessive pressure increases, regardless of their source. As discussed below, operability of the safety valves was demonstrated by prototypical testing and appropriate analyses.

18.2.5.4 Union Electric Response

The reactor coolant system is provided with two PORVs and three code safety valves. Each PORV also has an associated motor-operated block valve.

The PORVs for Callaway were manufactured by Crosby; the safety valves were manufactured by Crosby. These valves are included in the safety and relief valve testing program that has been developed by EPRI. A description of this program entitled "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," dated December 13, 1979, was submitted to the NRC on December 17, 1979 (letters from W. J. Cahill, Jr., Chairman of EPRI Safety and Analysis Task Force, to H. Denton and D. Eisenhut, NRC). A revision to this program was submitted to the NRC in July 1980. The NRC staff completed its review of this program and found it acceptable.

An interim report on these valve tests was submitted by the PWR utilities to the NRC in July 1981. A final report on these tests was provided in Reference 12.

Preoperational testing of the PORVs includes monitoring the dynamic response of the relief valve discharge piping during actuation of the PORVs. These in-plant dynamic tests will be initiated with a water-solid inlet (loop seal) at the PORVs and a steam bubble maintained in the pressurizer.

(4) For those designs in which instrument air is needed for operation, the electrical power supply should be required to have the capability to be manually connected to the emergency power sources.

18.2.14.2 Union Electric Response

The pressurizer level indication channels are powered from vital, Class 1E buses and displayed in the control room. These buses are described in Section 8.3; they are capable of being supplied from onsite emergency power (diesel generators) or offsite power.

The pressurizer PORVs and block valves are powered from vital, Class 1E power sources. The separation group assignment is indicated on system drawings in FSAR Section 5.1.

The pressurizer PORVs are relied on to perform ~~two~~ the following safety functions:

a) Pressure control during a shutdown concurrent with loss of offsite power;
b) Over pressure protection at low reactor coolant system pressures; and
c) RCS depressurization in the mitigation of the accidents discussed in Sections 15.5.1 and 15.6.3.

These functions are described in Sections 5.2 and 5.4 (A).

The PORV block valve is provided to isolate the PORV should the PORV develop an unacceptable leakage during operation.

The pressurizer level indication is used during normal operation to control pressurizer level (see Figure 7.2-1, sheet 11).

The pressurizer level indication is used for the reactor trip logic and is a displayed parameter for safe shutdown control. The safety design basis of the pressurizer level indication is provided in Section 7.2 and Section 7.5.

18.2.14.3 Conclusion

The Callaway design for the emergency power for pressurizer equipment satisfies Item II.G.1 of NUREG-0737. The Callaway design proposes an alternative to the power supply assignment proposed for the pressurizer PORVs and PORV block valves. The alternative is justified based on the diversity in power supply assignments for these valves, i.e., motor-operated (AC) block valves and solenoid-operated (DC) PORVs and based on the above requirements for PORV use. ~~for overpressure protection.~~

18.2.17 RECOMMENDATIONS FROM THE BULLETINS AND ORDERS TASK FORCE (II.K.3)

18.2.17.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System (II.K.3.1)

18.2.17.1.1 NRC Guidance Per NUREG-0737

Position

All PWR licensees should provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident. This system will automatically cause the block valve to close when the reactor coolant system pressure decays after the PORV has opened. Justification should be provided to ensure that failure of this system would not decrease overall safety by aggravating plant transients and accidents.

Each licensee shall perform a confirmatory test of the automatic block valve closure system following installation.

Clarification

Implementation of this action item was modified in the May 1980 version of NUREG-0660. The change delays implementation of this action item until after the studies specified in TMI Action Plan item II.K.3.2 have been completed, if such studies confirm that the subject system is necessary.

18.2.17.1.2 Union Electric Response

Westinghouse, as a part of the response prepared for the Westinghouse Owners Group to address item II.K.3.2 (refer to Section 18.2.17.2), has evaluated the necessity of incorporating an automatic pressurizer power-operated relief valve isolation system. This evaluation is documented in Reference 2 and concluded that such a system should not be required. ~~However,~~ The Callaway design includes the capability to ~~automatically~~ remote-manually isolate the power-operated relief valves ~~as discussed in Sections 7.6.6 and 7.6.10~~ by closing block valves from the main control room.

18.2.17.1.3 Conclusion

Based on the above discussion, Callaway meets the intent of the guidelines of NUREG-0737, Item II.K.3.1.

18.2.17.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System (II.K.3.2)

LOCA caused by a stuck-open PORV or safety valve. This analysis should consider modifications which have been made since the TMI-2 accident to improve the probability. This analysis shall evaluate the effect of an automatic PORV isolation system specified in Task Action Plan, Item II.K.3.1. In evaluating the automatic PORV isolation system, the potential of causing a subsequent stuck-open safety valve and the overall effect on safety (e.g., effect on other accidents) should be examined.

Actual operational data may be used in this analysis, where appropriate. The bases for any assumptions used should be clearly stated and justified.

The results of the probability analysis should then be used to determine whether the modifications already implemented have reduced the probability of a small-break LOCA due to a stuck-open PORV or safety valve a sufficient amount to satisfy the criterion stated above, or whether the automatic PORV isolation system specified in Task Action item II.K.3.1 is necessary.

In addition to the analysis described above, the licensee should compile operational data regarding pressurizer safety valves for PWR vendor designs. These data should then be used to determine safety-valve failure rates.

The analyses should be documented in a report. If this requirement is implemented on a generic basis, each licensee should review the appropriate generic report and document its applicability to his own plant(s). The report and the documentation of applicability (where appropriate) should be submitted for NRC staff review by the specified date.

18.2.17.2.2 Union Electric Response

As mentioned in item II.K.3.1 above (Section 18.2.17.1), the Westinghouse Owners Group has submitted a Westinghouse-prepared report (Ref. 2) which provides a probabilistic analysis to determine the probability of a PORV LOCA, estimates the effect of the post-TMI modifications, evaluates an automatic PORV isolation concept, and provides PORV and safety valve operational data for Westinghouse plants. Because of the sensitivity analyses included in the report, the report is generic and is applicable to Callaway. The report identifies a significant reduction in the PORV LOCA probability as a result of post-TMI modifications, and the calculations compare favorably with the operational data for Westinghouse plants (included as an appendix to the report).

and concludes that an automatic isolation capability is not required,

18.2.17.2.3 <u>Conclusion</u>

The requirements of this item were resolved by submittal of the analysis report discussed in Reference 2.

plants have proposed changing the derivative action setting to zero, thereby eliminating it from consideration. Either modification is acceptable to the staff. This represents a newly available option.

18.2.17.6.2 Union Electric Response

The Callaway design original included a pressure integral derivative (PID) controller in the power-operated relief valve control circuit. The time derivative constant in the PID controller for the pressurizer PORV was turned to "OFF" prior to commercial operation. The appropriate plant procedure for calibrating the set points in this ~~nonsafety grade~~ system reflected this decision.

Setting the derivative time constant to "OFF," in effect, removed the derivative action from the controller. Removal of the derivative action decreased the likelihood of opening the pressurizer PORV since the actuation signal for the valve was then no longer sensitive to the rate of change of pressurizer pressure. This PID controller was used in the ~~control~~ actuation circuitry for BB-PCV-0455A. A subsequent plant modification revised the PORV ~~control~~ actuation circuitry such that the PID controller is no longer used in the ~~control~~ actuation circuitry for BB-PCV-0455A. The current design is reflected in Figures 7.7-4 and 7.2-1, sheet 11.

18.2.17.6.3 <u>Conclusion</u>

The NUREG-0737 provisions for the PID controller no longer apply to Callaway.

18.2.17.7 <u>Proposed Anticipatory Trip Modification (II.K.3.10)</u>

18.2.17.7.1 NRC Guidance Per NUREG-0737

Position

The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power-operated relief valve (PORV) is substantially unaffected by the modification.

Clarification

This evaluation is required for only those licensees/applicants who propose the modification.

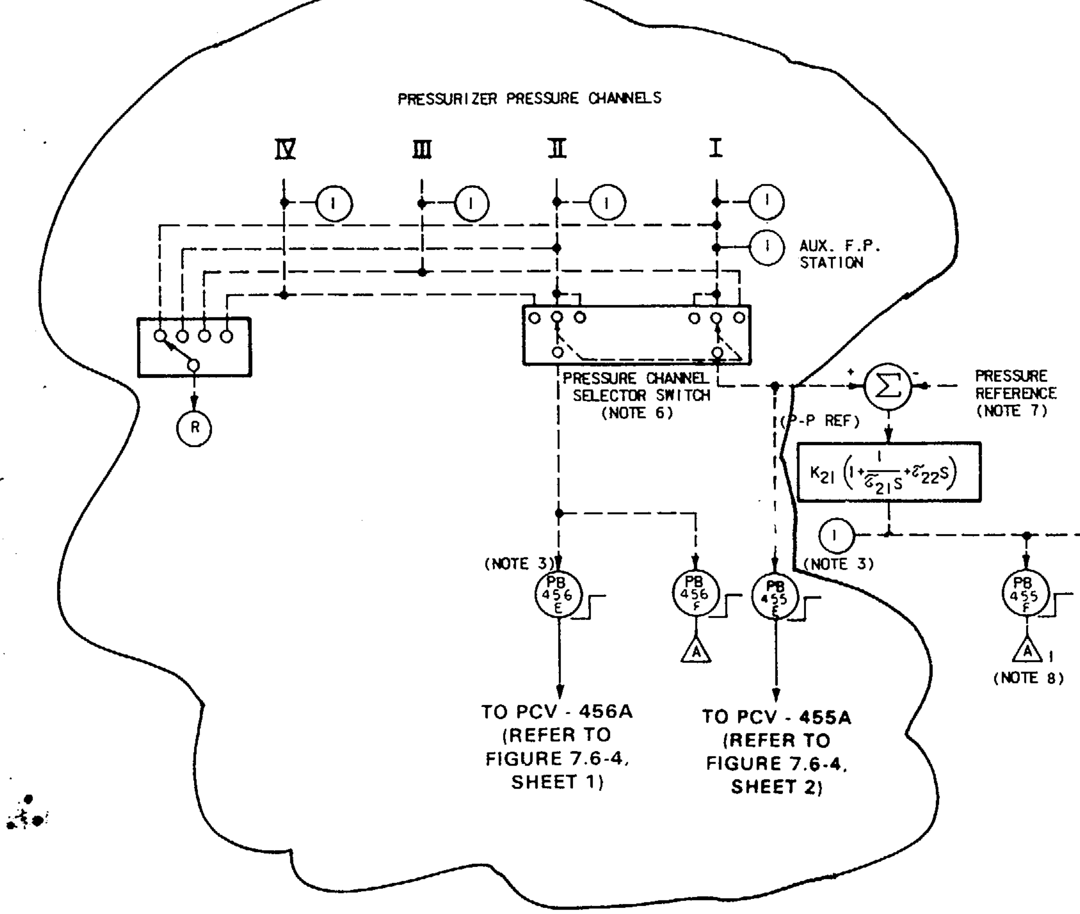
18.2.17.7.2 Union Electric Response

This anticipatory trip modification is included in the SNUPPS design.

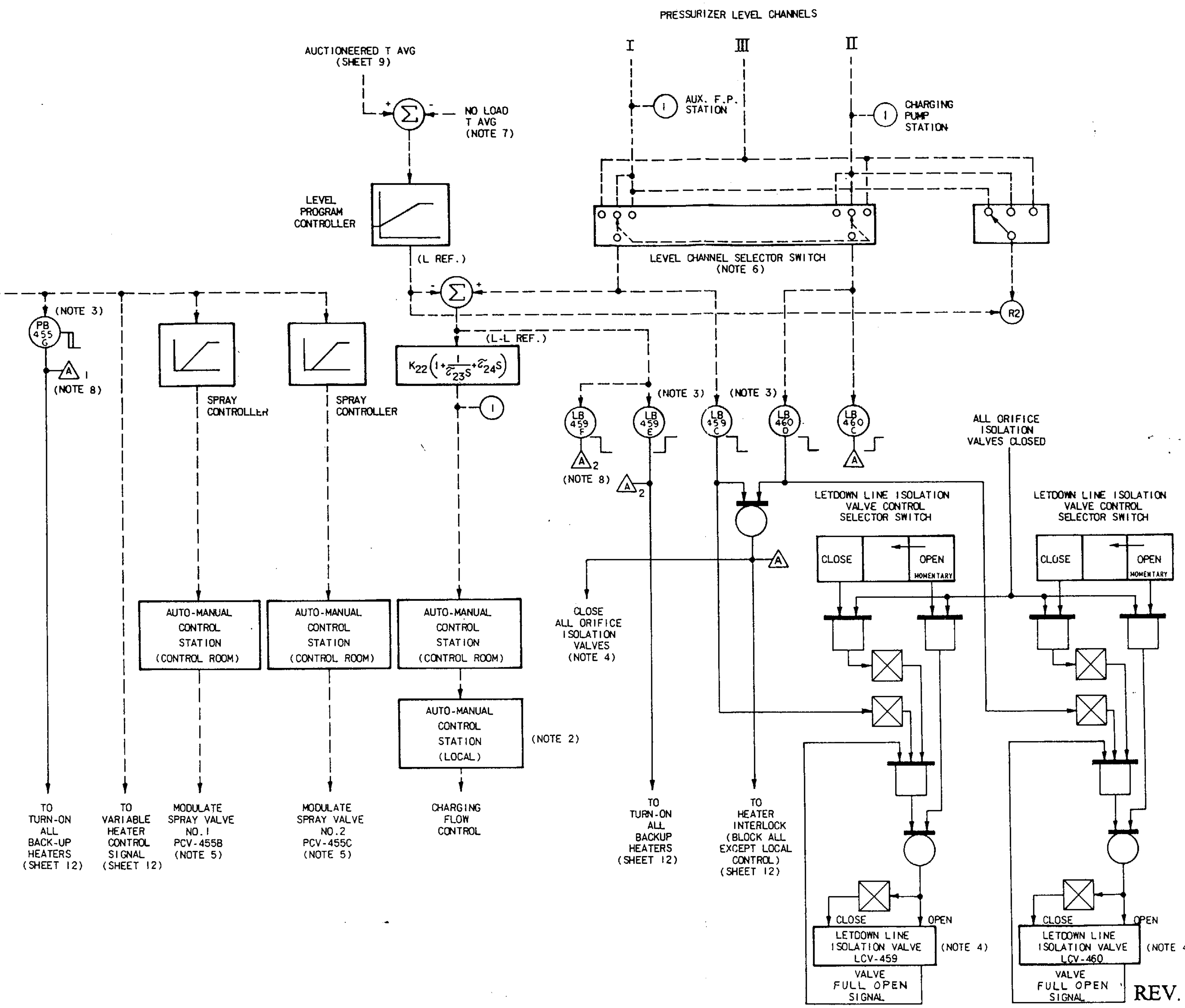
The NRC has raised the question of whether the pressurizer power-operated relief valves would be actuated for a turbine

ATTACHMENT SIX

PORV CONTROL SCHEMATICS



(will be modified to fully Class IE)



NOTES:

1. ALL CIRCUITS ON THIS SHEET ARE NOT REDUNDANT.
2. LOCAL CONTROL OVERRIDES ALL OTHER SIGNALS. LOCAL OVERRIDE ACTUATES ALARM IN CONTROL ROOM.
3. PRESSURE BISTABLES NO. PB-455E, PB-455G, PB-456E, PB-457E, & PB-4588 AND LEVEL BISTABLES NO. LB-459C, LB-459E, & LB-4600 ARE ENERGIZE TO ACTUATE.
4. OPEN/SHUT INDICATION IN CONTROL ROOM.
5. A LIGHT IS PROVIDED IN THE CONTROL ROOM FOR EACH SPRAY VALVE TO INDICATE WHEN THE VALVE IS CLOSED.
6. CENTER POSITION NORMALLY SELECTED.
7. ADJUSTABLE SETPOINT WITHIN CONTROLLER.
8. ALARM 1 AND ALARM 2 HAVE REFLASH CAPABILITY.

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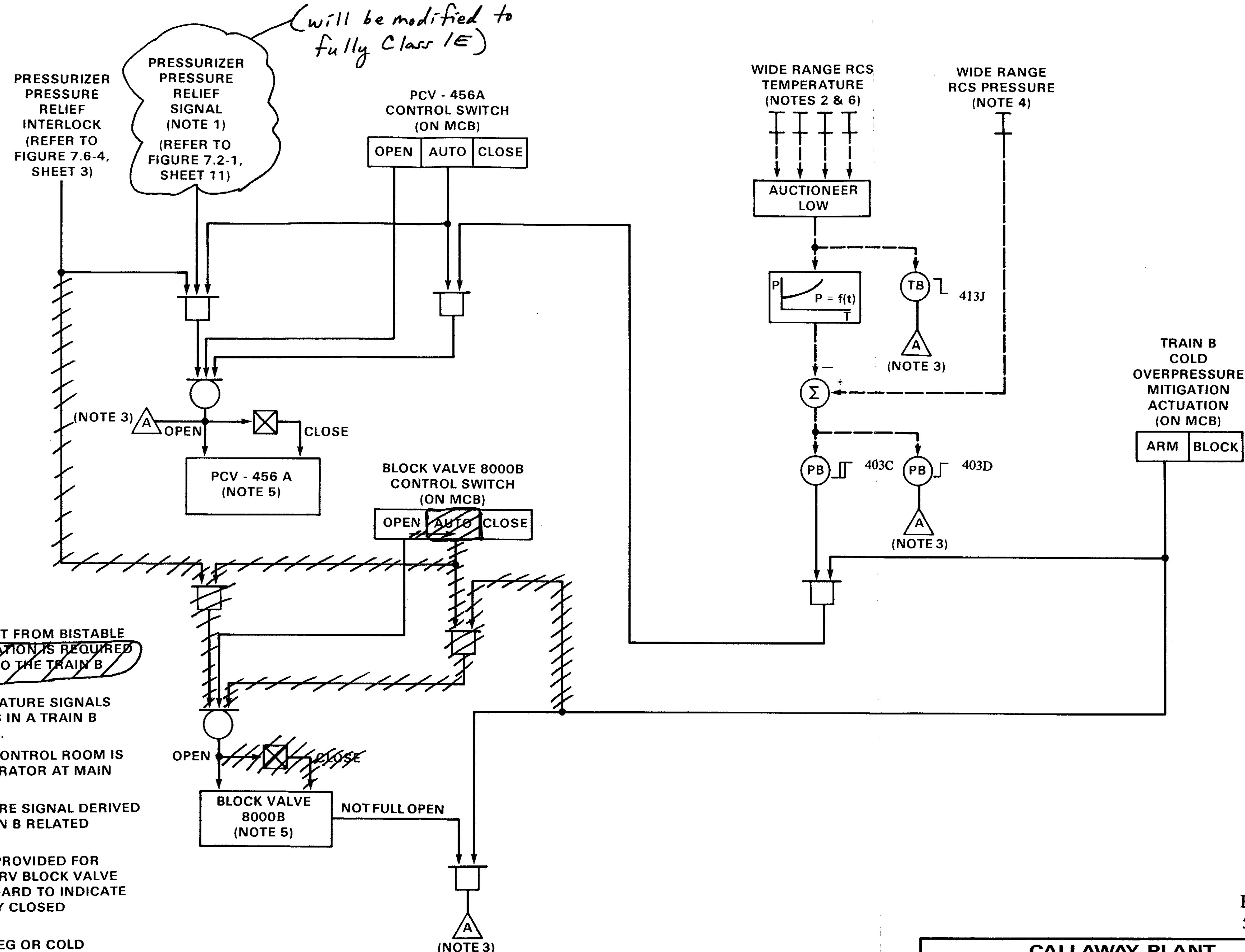
CALLAWAY PLANT

FIGURE 7.2-1
FUNCTIONAL DIAGRAMS
(PRESSURIZER PRESSURE AND LEVEL CONTROL)
(SHEET 11)

19,943.2



NOTES:

1. THIS SIGNAL IS THE OUTPUT FROM BISTABLE PB-456E. ~~ELECTRICAL ISOLATION IS REQUIRED TO BRING THIS SIGNAL INTO THE TRAIN B PROTECTION CABINET.~~
2. WIDE RANGE RCS TEMPERATURE SIGNALS DERIVED FROM CHANNELS IN A TRAIN B RELATED PROTECTION SET.
3. ANNUNCIATION IN MAIN CONTROL ROOM IS REQUIRED VISIBLE TO OPERATOR AT MAIN CONTROL BOARD.
4. WIDE RANGE RCS PRESSURE SIGNAL DERIVED FROM CHANNEL IN A TRAIN B RELATED PROTECTION SET.
5. STATUS LIGHTS MUST BE PROVIDED FOR EACH PORV AND EACH PORV BLOCK VALVE AT THE MAIN CONTROL BOARD TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
6. THE RCS LOOP AND HOT LEG OR COLD LEG ASSIGNMENTS FOR THE WIDE RANGE RCS TEMPERATURE SIGNALS MUST BE CONSISTENT WITH THE REQUIREMENTS FOR RVLIS AND PAMS.

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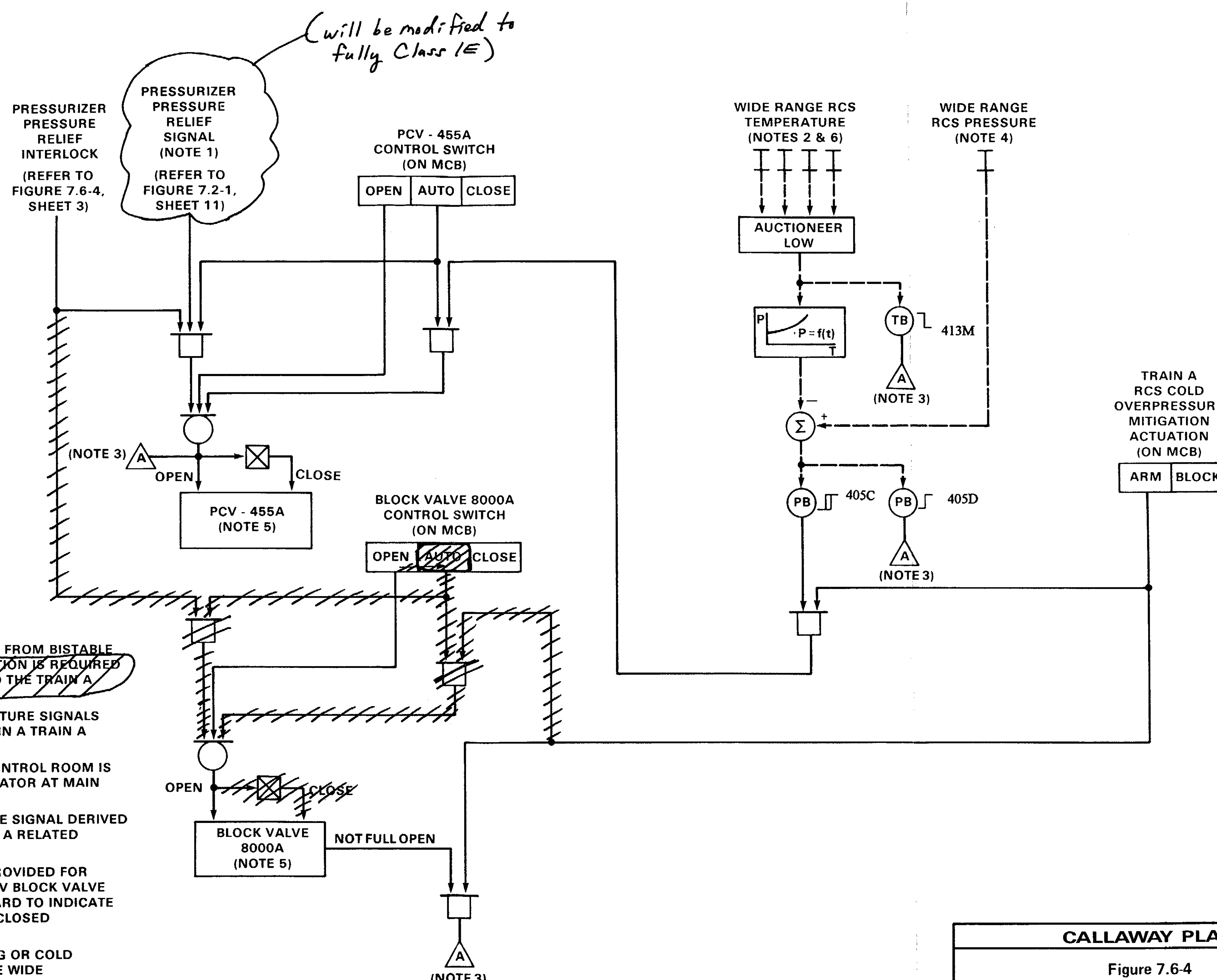
5/94

CALLAWAY PLANT

Figure 7.6-4

Train B Functional Diagram Showing Logic Requirements for Pressurizer Pressure Relief System
Sheet 1



NOTES:

1. ~~THIS SIGNAL IS THE OUTPUT FROM BISTABLE PB-455E. ELECTRICAL ISOLATION IS REQUIRED TO BRING THIS SIGNAL INTO THE TRAIN A PROTECTION CABINET.~~
2. WIDE RANGE RCS TEMPERATURE SIGNALS DERIVED FROM CHANNELS IN A TRAIN A RELATED PROTECTION SET.
3. ANNUNCIATION IN MAIN CONTROL ROOM IS REQUIRED VISIBLE TO OPERATOR AT MAIN CONTROL BOARD.
4. WIDE RANGE RCS PRESSURE SIGNAL DERIVED FROM CHANNEL IN A TRAIN A RELATED PROTECTION SET.
5. STATUS LIGHTS MUST BE PROVIDED FOR EACH PORV AND EACH PORV BLOCK VALVE AT THE MAIN CONTROL BOARD TO INDICATE WHEN THE VALVE IS FULLY CLOSED OR FULLY OPEN.
6. THE RCS LOOP AND HOT LEG OR COLD LEG ASSIGNMENTS FOR THE WIDE RANGE RCS TEMPERATURE SIGNALS MUST BE CONSISTENT WITH THE REQUIREMENTS FOR RVLIS AND PAMS.

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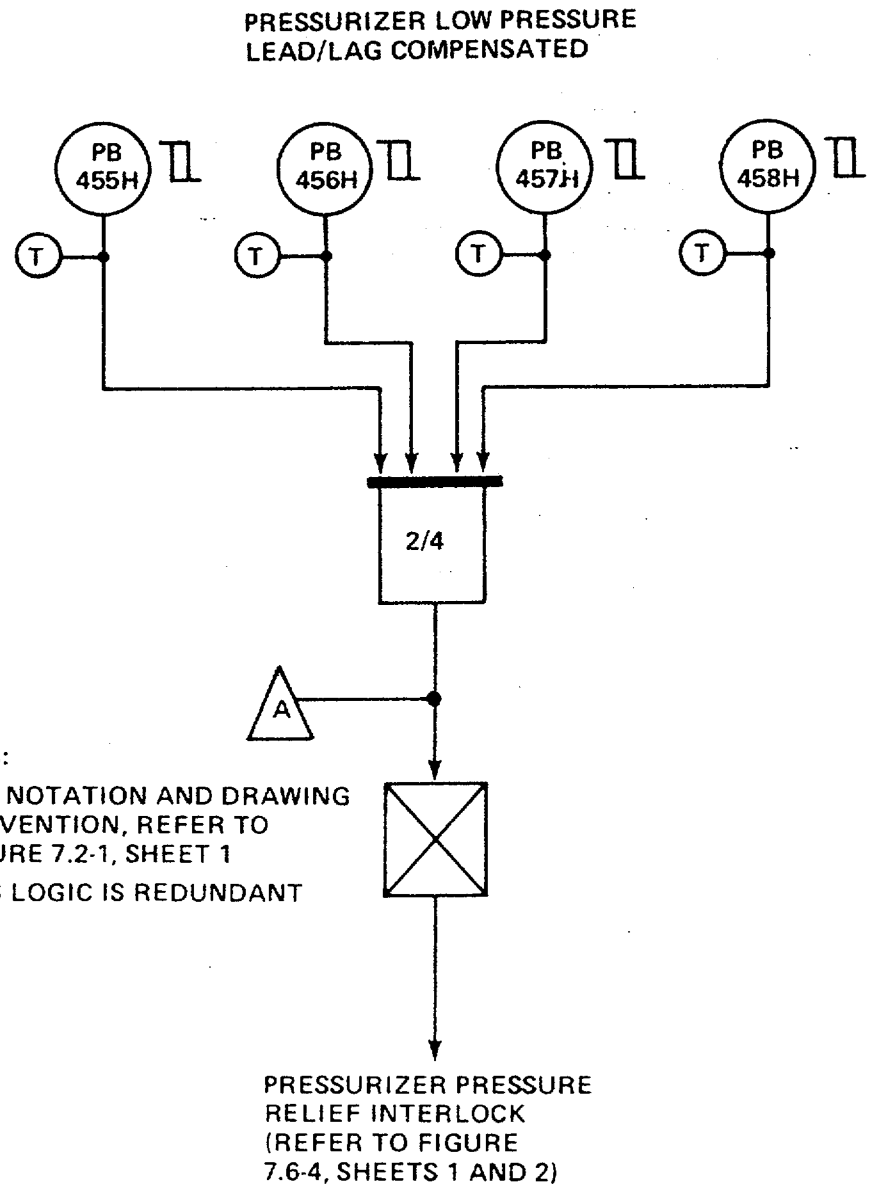
5/94

CALLAWAY PLANT

Figure 7.6-4

Train A Functional Diagram Showing Logic Requirements for Pressurizer Pressure Relief System

Sheet 2



NOTES:

1. FOR NOTATION AND DRAWING CONVENTION, REFER TO FIGURE 7.2-1, SHEET 1
2. THIS LOGIC IS REDUNDANT

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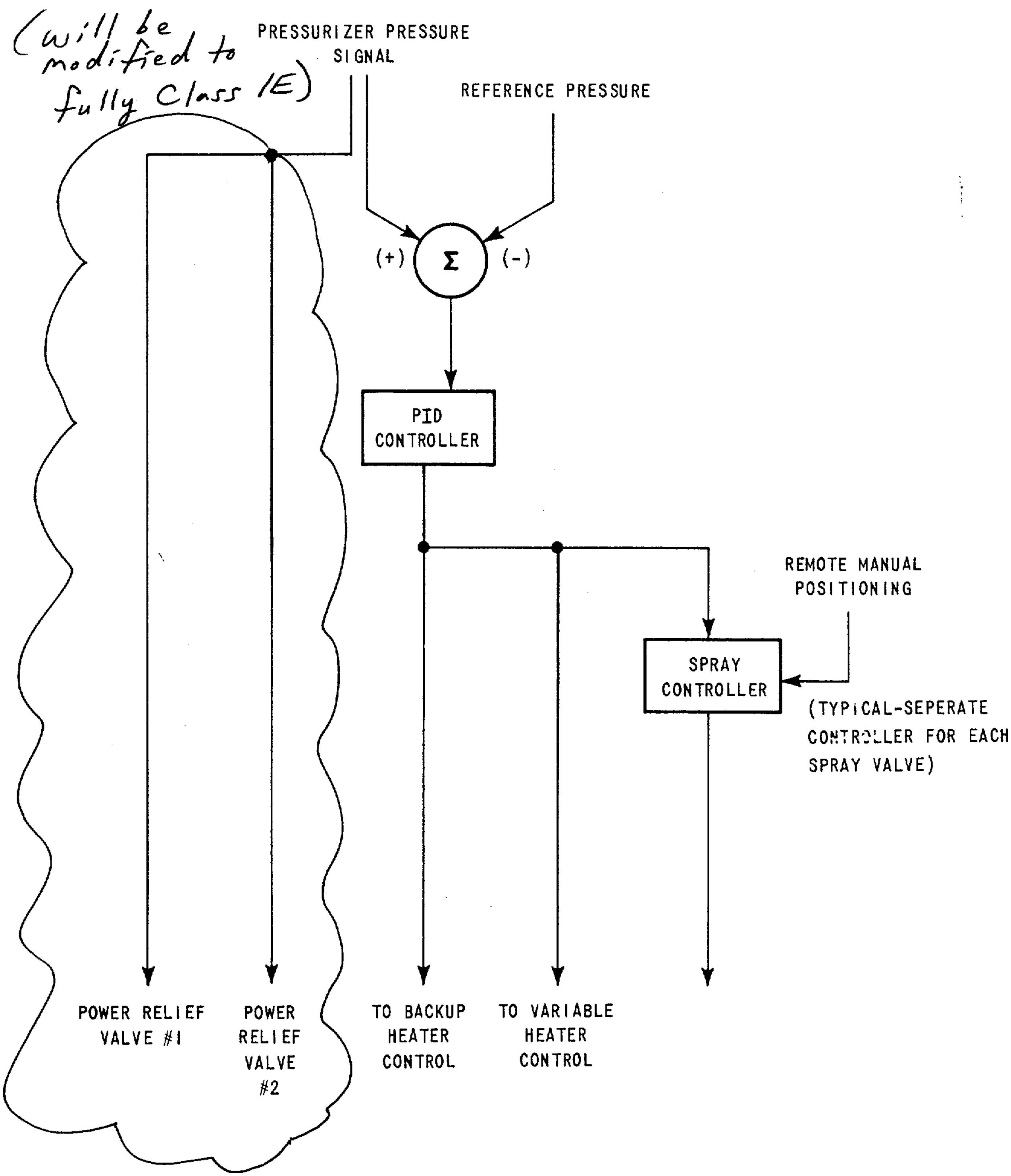
6/86

CALLAWAY PLANT

Figure 7.6-4

Functional Diagram of Logic Requirement for Pressurizer Pressure Relief System Interlock
Sheet 3



PID - PROPORTIONAL + INTEGRAL + DERIVATIVE

REV. OL-7
5/94

CALLAWAY PLANT

FIGURE 7.7-4
BLOCK DIAGRAM OF PRESSURIZER PRESSURE CONTROL SYSTEM

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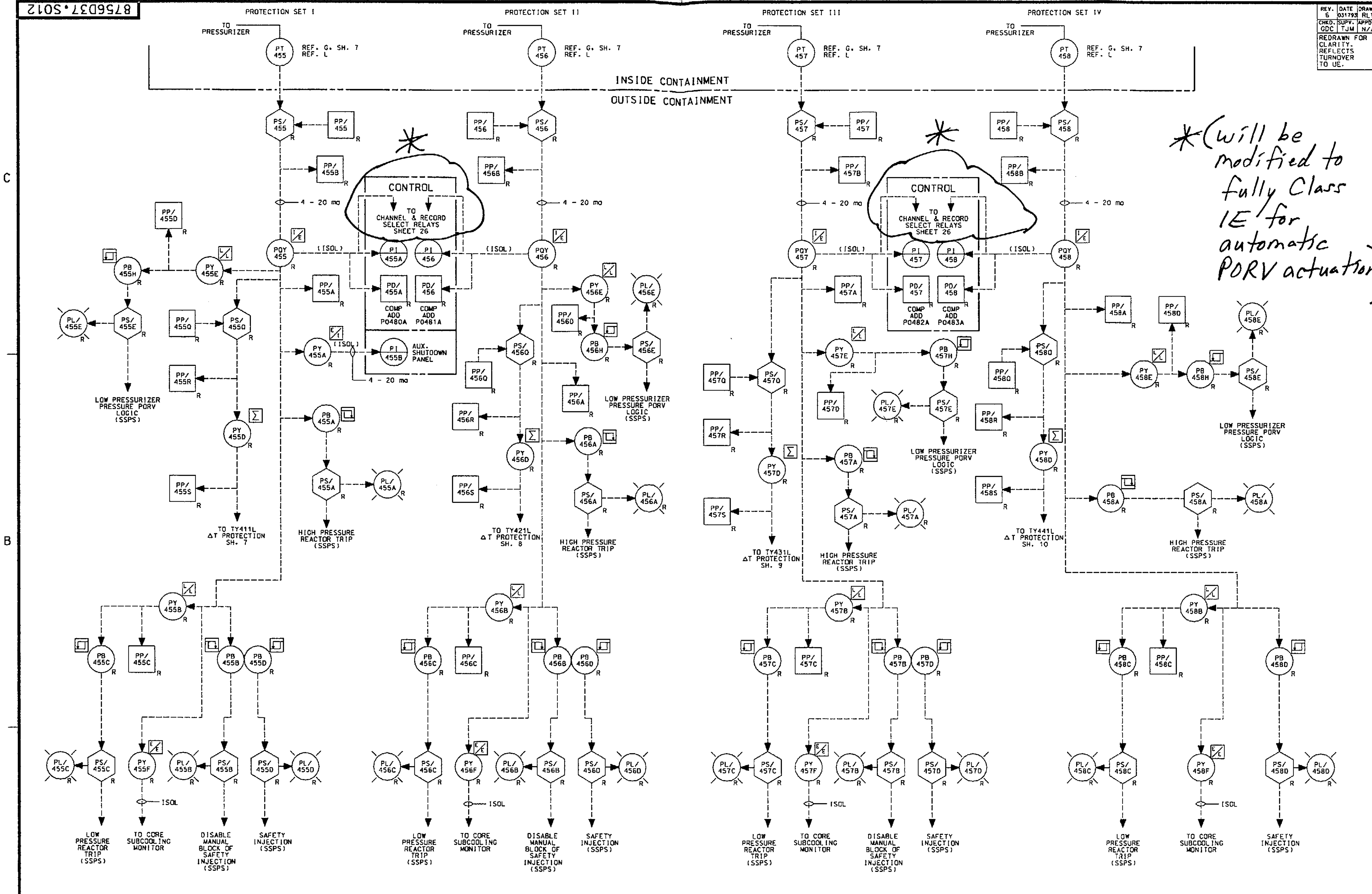


*(will be modified to fully Class IE for automatic PORV actuation)

REV.	DATE	DRAWN
6	031793	RLW
CHKD.	SUPV.	APPD.
GDC	TJM	N/A

REDRAWN FOR CLARITY. REFLECTS TURNOVER TO UE.

NOTES:

SEE SHEET 1 FOR NOTES AND REFERENCES.

CLASS 1

~~M-761-00077~~ SYS.: SB. SC

STANDARDIZED NUCLEAR UNIT POWER PLANT SYSTEM PROJECTS
UNIT: SNP ITEM-10
STATUS: CERTIFIED FOR CONSTRUCTION
CERTIFICATION LTR. NO. SNP-4981
AUTHORITY: J.B. COOKINHAM
ENGR. LTR. NO. EP/SA-38699

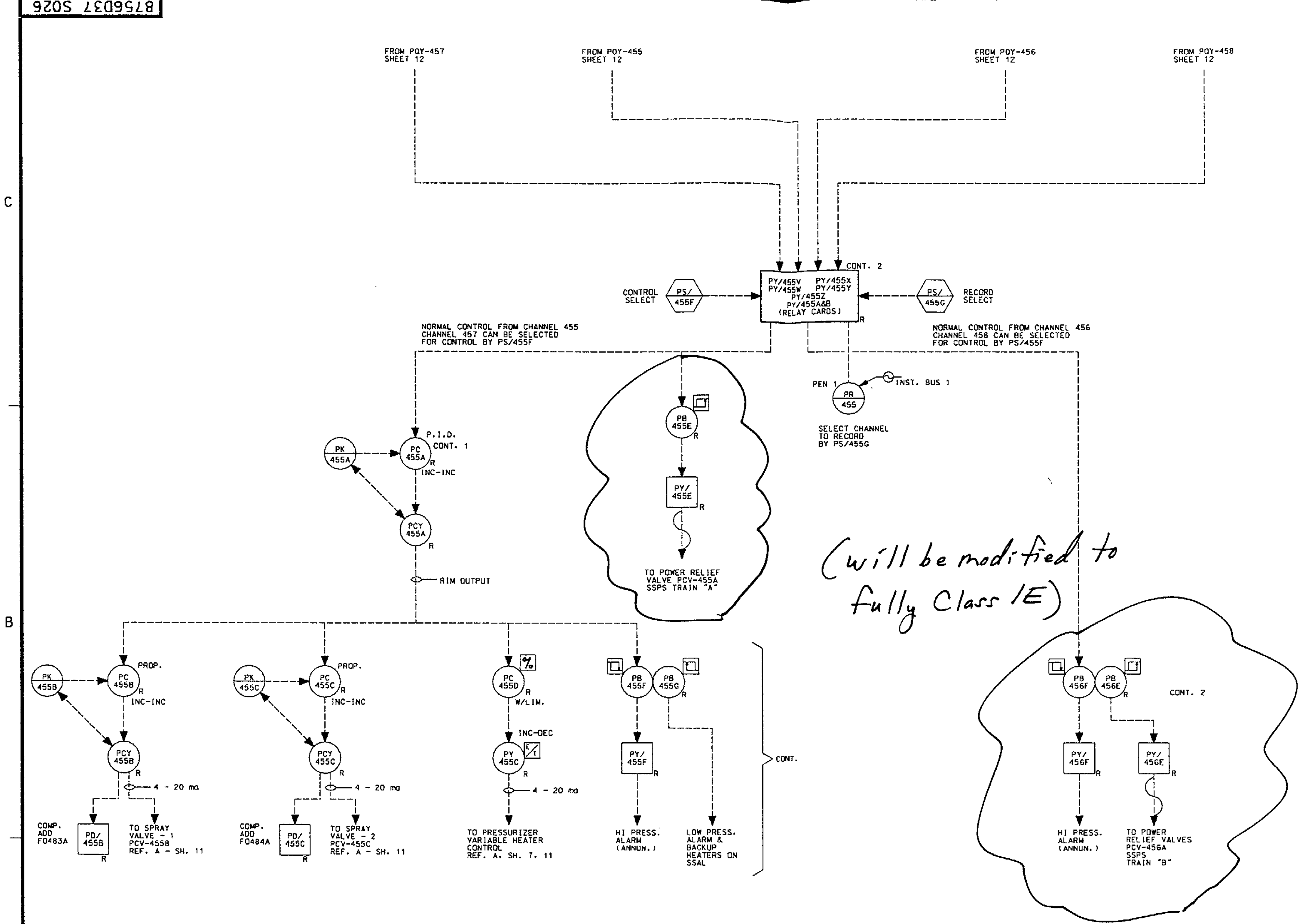
DRAWN N/A
CHKD. N/A
SUPV. N/A
APPR. N/A

PROCESS CONTROL BLOCK DIAGRAM
PRESSURIZER PRESSURE
PROTECTION SETS I, II, III, IV

CALLAWAY PLANT

UNION ELECTRIC COMPANY
ST. LOUIS, MO

8756D37.S012 REV. 6



8756D37 S026

REV.	DATE	DRAWN
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CHKD.	SUPV.	APPD.
GDC	TJM	N/A
REDRAWN FOR CLARITY.		

NOTES:

SEE SHEET 1 FOR NOTES & REFERENCES.

CLASS 1

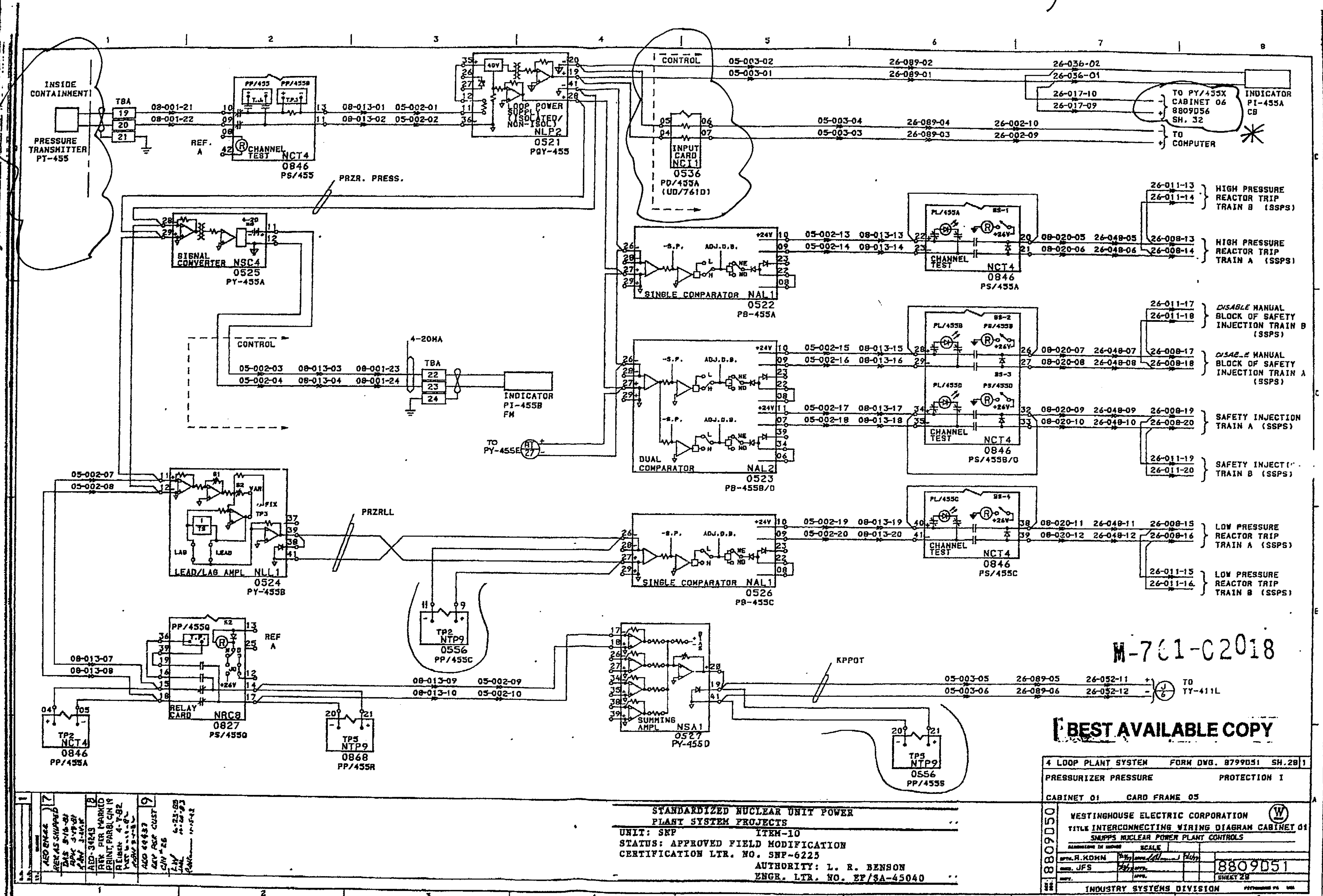
~~M-761-00091~~ SYS: SB, SC

STANDARDIZED NUCLEAR UNIT POWER PLANT SYSTEM PROJECTS

UNIT: SNP ITEM-10
STATUS: CERTIFIED FOR CONSTRUCTION
CERTIFICATION LTR. NO. SNP-5463
AUTHORITY: J.B. COOKINGHAM
ENGR. LTR. NO. EP/SA-41407

DRAWN (DATE) N/A	PROCESS CONTROL BLOCK DIAGRAM PRESSURIZER PRESSURE CONTROL
CHKD. (DATE) N/A	
SUPV. (DATE) N/A	
APPD. (DATE) N/A	LOCATION CALLAWAY PLANT CLASS
UNION ELECTRIC COMPANY ST. LOUIS, MO	8756D37 S026 REV. 9

*(will be modified to fully Class IE for automatic PORV actuation)



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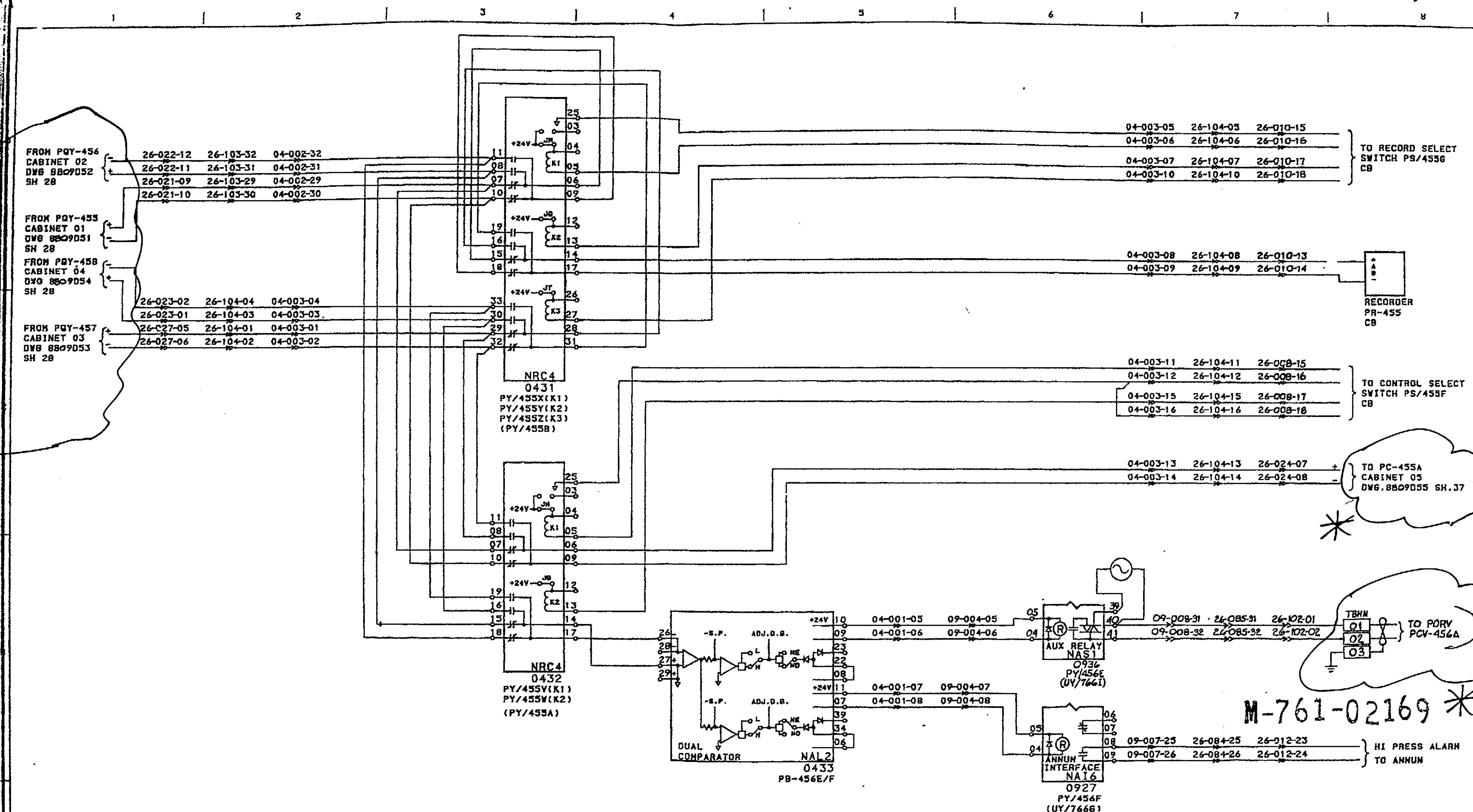

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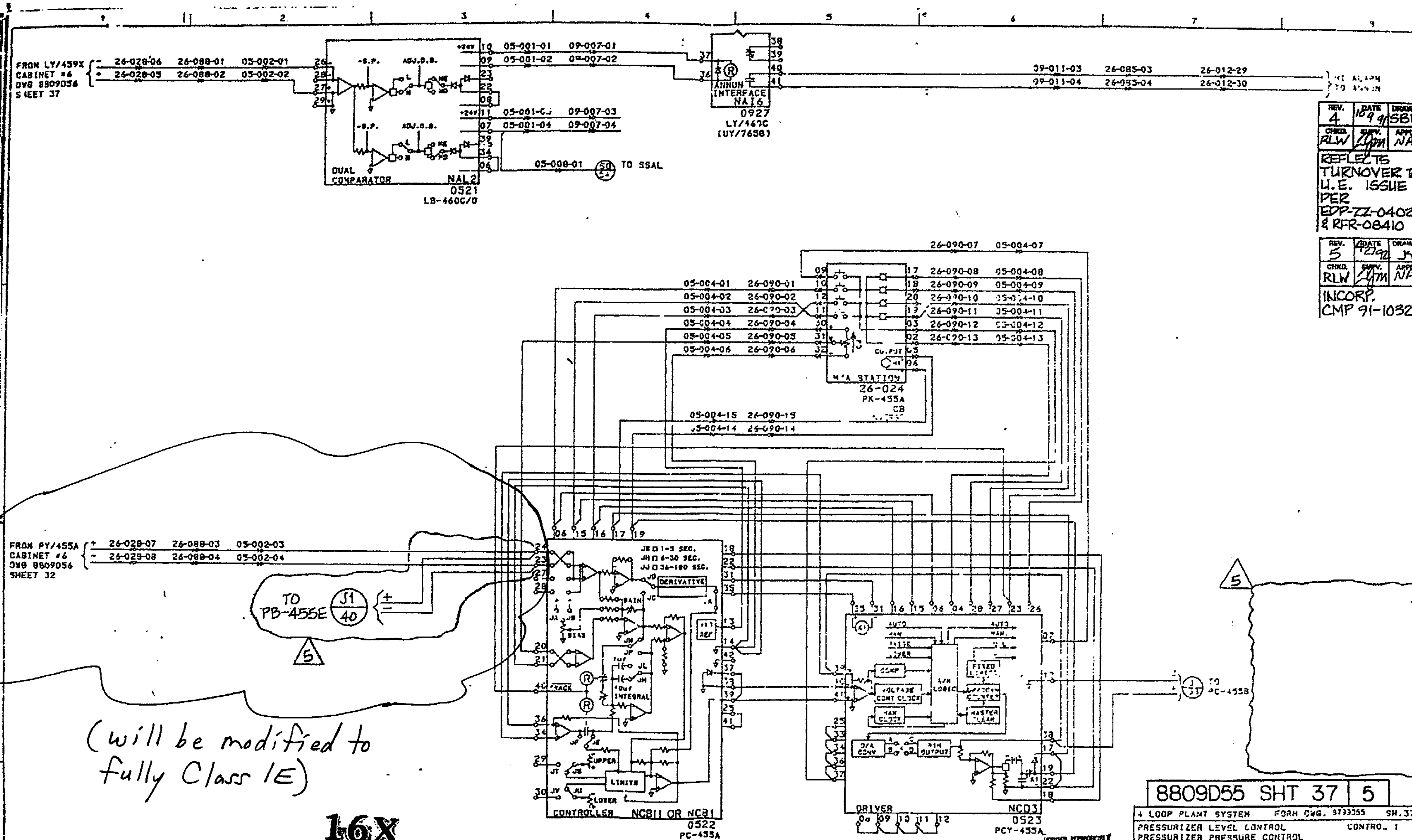
STANDARDIZED NUCLEAR UNIT POWER PLANT SYSTEM PROJECTS
UNIT: SNP ITEM-10
STATUS: APPROVED FIELD MODIFICATION
CERTIFICATION LTR. NO. SNP-6225
AUTHORITY: L. R. BENSON
ENGR. LTR. NO. EF/SA-45040

4 LOOP PLANT SYSTEM FORM DWG. 8799036 SH.32
PRESSURIZER PRESSURE CONTROL CONTROL 2
CABINET 06 CARD FRAME 04
WESTINGHOUSE ELECTRIC CORPORATION
TITLE INTERCONNECTING WIRING DIAGRAM CABINET 06
SNUPPS NUCLEAR POWER PLANT CONTROLS
8809D56
SHEET 32
INDUSTRY SYSTEMS DIVISION



REV.	DATE	DRAWN
4	10/9/91	SBR
CHKD. RLW	SUPV.	APPD. NA

REFLECTS TURNOVER TO W.E. ISSUE PER EDP-ZZ-04024 & RFR-08410

REV.	DATE	DRAWN
5	4/21/92	
CHKD. RLW	SUPV.	APPD. NA

INCORP. CMP 91-1032.

STANDARDIZED NUCLEAR UNIT POWER PLANT SYSTEM PROJECTS
ITEM 10
UNITS: SAP/SCP
STATUS: AS MANUFACTURED
CERTIFICATION LETTER NO. SNP-3055
AUTHORITY: A. F. JOST
ENGR. LTR. NO. EP/SA-27655

8809D55 SHT 37 | 5

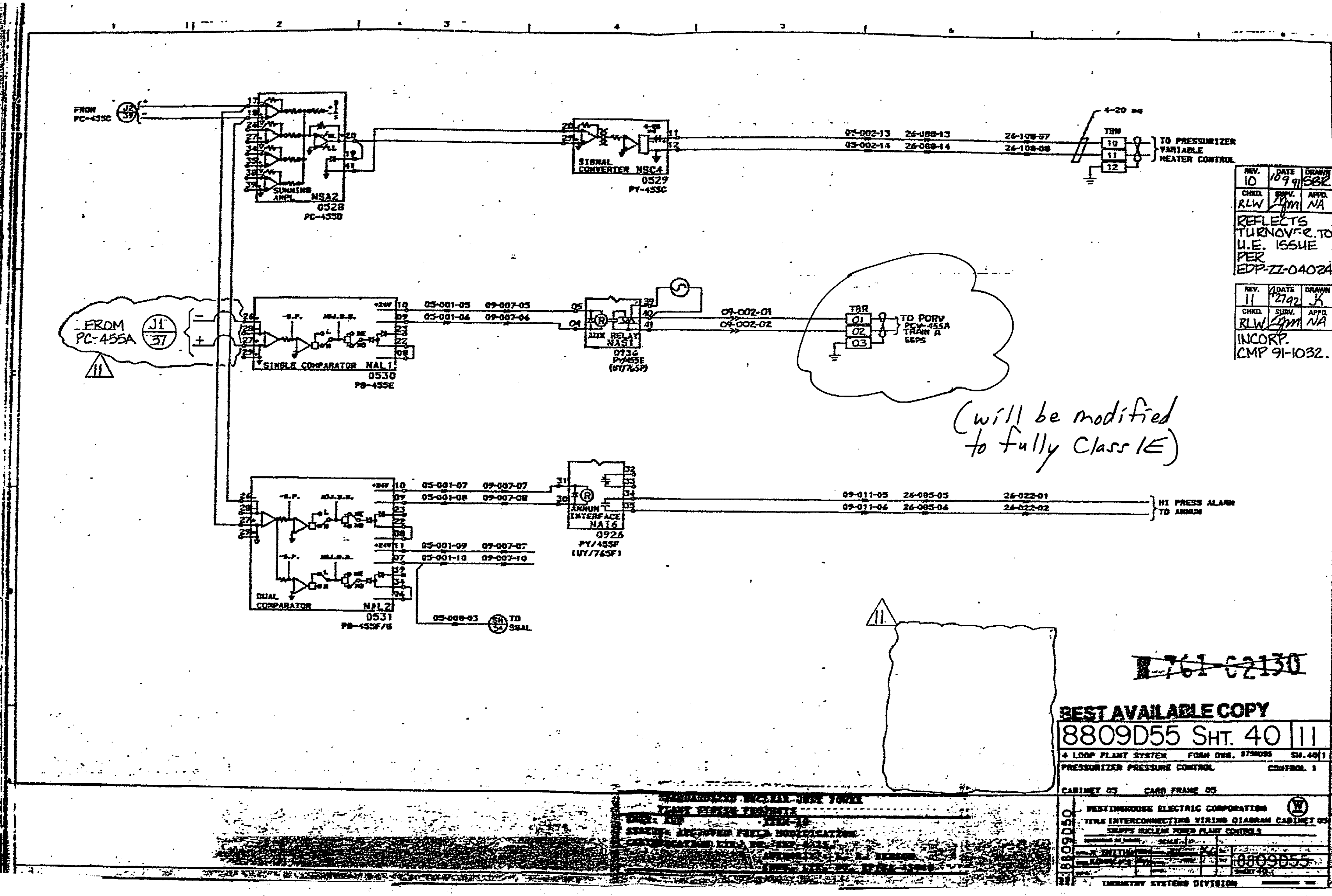
4 LOOP PLANT SYSTEM FORM DWG. 9733355 SH.37
PRESSURIZER LEVEL CONTROL CONTROL 1
PRESSURIZER PRESSURE CONTROL
CABINET 05 CARD FRAME 05

WESTINGHOUSE ELECTRIC CORPORATION
TITLE INTERCONNECTING WIRING DIAGRAM CABINET
SNUPPS NUCLEAR POWER PLANT CONTROLS

8809D55
SHEET 37
INDUSTRY SYSTEMS DIVISION

I CERTIFY THAT THE IMAGE CONTAINED ON THIS FRAME WAS MADE IN THE NORMAL AND REGULAR COURSE OF BUSINESS, ON THE DATE STATED BELOW AND THAT IT IS AN ACCURATE REPRODUCTION OF THE DOCUMENT(S) SUBMITTED TO MICROGRAPHICS.

DATE 4-28-92 OPERATOR P. Powell SUPERVISOR Grace Chamley



FROM PC-455C
SUMMING AMPL. NSA2
0528
PC-455D
SIGNAL CONVERTER NSC4
0529
PY-455C
4-20 ma
TBN
TO PRESSURIZER VARIABLE HEATER CONTROL
FROM PC-455A
J1
37
SINGLE COMPARATOR NAL1
0530
PB-455E
AUX RELAY NAS1
0736
PY455E
TBR
TO PORV PCV-455A TRAIN A SEPS
(will be modified to fully Class IE)
DUAL COMPARATOR NAL2
0531
PB-455F/G
ANNUN INTERFACE NAI6
0926
PY/455F
TO SEAL
HI PRESS ALARM TO ANNUN
REV. 10 DATE 10/9/91 DRAWN SBR
CHKD. RLW APPD. NA
REFLECTS TURNOVER TO U.E. ISSUE PER EDP-ZZ-04024
REV. 11 DATE 1/27/92 DRAWN JK
CHKD. RLW APPD. NA
INCORP. CMP 91-1032.
~~E-761-C2130~~
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8809D55 SHT. 40 11
4 LOOP PLANT SYSTEM
PRESSURIZER PRESSURE CONTROL
CONTROL 1
CABINET 05 CARD FRAME 05
WESTINGHOUSE ELECTRIC CORPORATION
TITLE INTERCONNECTING WIRING DIAGRAM CABINET 05
SNUPPS NUCLEAR POWER PLANT CONTROLS
8809D55
INDUSTRY SYSTEMS DIVISION