Docket File 50-423

DEC 15 1983

MEMORANDUM FOR: Thomas M. Novak, Assistant Director for Licensing, DL

FROM: R. Wayne Houston, Assistant Director for Reactor Safety, DSI

SUBJECT: ICSB INPUT TO DRAFT SER - MILLSTONE UNIT 3

Plant Name: Millstone Unit 3
Docket No.: 50-423
Licensing Stage: OL
Responsible Branch: LB #1
Project Manager: E. Doolittle
Review Branch: ICSB
Review Status: Incomplete

DESIGNATED ORIGINAL
Certified By [signature]

Enclosed is a draft Safety Evaluation Report (SER) prepared by the Instrumentation and Control Systems Branch (ICSB). This draft SER contains the results of our review of the information presented in the Millstone Unit 3 Final Safety Analysis Report (FSAR) through Amendment No. 5. Also, the draft SER is based on a drawing review and our evaluation of information provided by the applicant during ICSB review meetings which were held on July 26-28, 1983, and December 1, 1983.

The open items in the Instrumentation and Control System review for Millstone Unit 3 are identified in Section 7.1.4 of the enclosed draft SER. There are 22 open items and 11 confirmatory items identified at the present time. Our continuing review may lead to new open issues in addition to those described in this draft. Any new open issues will be forwarded to the project manager for follow-up action with the applicant. The evaluation conclusion for each section will be addressed in the final SER.

Original signed by
R. Wayne Houston

R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

Enclosure:
As stated

cc: R. Mattson
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ENCLOSURE 1

DRAFT SAFETY EVALUATION REPORT FOR MILLSTONE UNIT NO. 3

DESIGNATED ORIGINAL

Certified By Cheryl Thompson

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.1 Acceptance Criteria

FSAR Section 7.1 contains information pertaining to safety-related instrumentation and control systems, their design bases, and applicable acceptance criteria. The staff has reviewed the applicant's design, design criteria, and design bases for the instrumentation and control systems for Millstone Unit 3. The acceptance criteria used as the basis for this evaluation are those identified in the SRP (NUREG-0800) in Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems Important to Safety," and Table 7-2, "TMI Action Plan Requirements for Instrumentation and Control System Important to Safety." These acceptance criteria include the applicable GDC and the Institute of Electrical and Electronics Engineers (IEEE) Standard 279 "Criteria for Protection System for Nuclear Power Generating Stations" (10 CFR 50.55a(h)). Guidelines for implementation of the requirements of the acceptance criteria are provided in IEEE standards, RGs, and BTPs identified in SRP

7.1. Conformance to the acceptance criteria provides the bases for concluding that the instrumentation and control systems meet the requirements of 10 CFR 50.

7.1.2 Method of Review

Millstone Unit 3 uses a Westinghouse NSSS with balance-of-plant (BOP) design provided by Stone and Webster Engineering Corporation. Many safety-related instrumentation and control systems are similar to those at Comanche Peak or McGuire and have been previously reviewed and approved by the staff. The staff concentrated its review on those areas where the Millstone Unit 3 design differs from previously reviewed designs and on those areas that have remained of concern during reviews of other similar plants. Several meetings were held with the applicant and the NSSS and BOP designers to clarify the design and to discuss staff concerns. Detail Drawings--including piping and instrumentation diagrams, logic diagrams, control wiring diagrams, electrical one-line diagrams, and electrical schematic diagrams--were audited during the review.

7.1.3 General Conclusion

The applicant has identified the instrumentation and control systems important to safety and the acceptance criteria that are applicable to those systems as identified in the SRP. The applicant has also identified the guidelines--including the RGs and the industry codes and standards--that are applicable to the systems as identified in FSAR Table 7.1-1.

Based on the review of FSAR Section 7.1, the staff concludes that the implementation of the identified acceptance criteria and guidelines satisfies the requirements of GDC 1, "Quality Standards and Records", with respect to the design fabrication, erection, and testing to quality standards commensurate with the importance of the safety functions to be performed. The staff finds that the NSSS and the BOP instrumentation and control systems important to safety, addressed in FSAR Section 7.1, satisfy the requirements of GDC 1 and, therefore, are acceptable.

7.1.4 Specific Findings

7.1.4.1 Open Items

The staff's conclusions apply to the instrumentation and control systems important to safety with the exception of the open items listed below. The staff will review these items and report their resolution in ~~a subsequent~~ the final version of this report. The applicable sections of this report that address these items are indicated in parentheses following each open item.

1. Design modification for automatic reactor trip using shunt coil trip attachment (7.2.2.4)

2. Conformance with Branch Technical Position ICSB-26 (7.2.2.7)

3. Steam generator level control and protection (7.3.3.4)

4. Containment isolation for the main steam lines to the turbine of the AFW pump (7.3.3.6)

5. Letdown line relief valve (7.3.3.7)

6. Non-Class 1E control signals to Class 1E control circuits (7.3.3.11)

7. Isolators used in the BOP design for isolation between safety and nonsafety-related systems (7.3.3.12)

8. Feedwater isolation and control valves (7.3.3.13)

9. BOP instrumentation and control system testing capability (7.3.3.15)

10. Remote shutdown capability (7.4.2.3)

11. IE Bulletin 79-27 concerns (7.5.2.1)

12. Bypass and inoperable status panel (7.5.2.2)

13. NUREG-0737 Item II.F.1 Accident Monitoring Instrumentation Position (4), (5), and (6).

14. NUREG-0737 Item II.F.2 - Instrumentation for Detection of Inadequate Core Cooling (7.5.2.5)

15. Emergency Response Capability - R.G. 1.97 Rev. 2 requirements (7.5.2.6)

16. RHR system isolation valve interlock (7.6.2.1)

17 Isolation of low pressure systems from the high pressure RCS (7.6.2.2)

18. RCS overpressure protection (7.6.2.3)

19. Reactor coolant system loop isolation valve interlocks (7.6.2.5)

20. Control system failures caused by malfunctions of common power source or instrument line (7.7.2.1)

21. Control system failures caused by high energy line breaks (7.7.2.2)

22. Freeze protection for instrument sensing lines (7.7.2.3) .

7.1.4.2 Confirmatory Items

In a number of cases, the applicant has committed to provide additonal documentation to address concerns raised by the staff during its review. Based on information provided during meetings and discussions with the applicant, the technical issue has been resolved in an acceptable manner. However, the applicant must formally document his commitments for resolution of these items. The sections of this report that address these items are indicated in parentheses.

(1) Cable separation in NSSS process cabinets (7.2.2.1)

(2) Reactor coolant pump underspeed trip (7.2.2.6)

(3) Test of Engineered Safeguard P-4 interlock (7.3.3.2)

(4) IE Bulletin 80-06 concerns (7.3.3.5)

(5) Control building isolation reset (7.3.3.8)

(6) Power lockout feature for motor operated valves (7.3.3.9)

(7) Failure mode and effects analyses of ESFAS (7.3.3.10)

(8) Sequencer deficiency report (7.3.3.14)

(11) Accumulator isolation valve interlock (7.6.2.4)

7.1.4.3 Technical Specification Items

Items to be included in the plant Technical Specifications and information to be audited as part of the review of proposed Technical Specifications are discussed in the following sections:

1. Trip setpoint and margins (7.2.2.2)
2. Response-time testing (7.2.2.3)
3. Spare component cooling water pump (7.3.3.16)

7.1.4.4 Site Visit

A site review will be performed to confirm that the physical arrangement and installation of electrical equipment are in accordance with the design criteria and descriptive information reviewed by the staff. The site review will be completed before a license is issued; any problems found will be addressed in a supplement to this report.

7.1.4.5 Fire Protection Review

The review of the auxiliary shutdown panel discussed in Section 7.4 of this report includes the compliance of this panel with GDC 19, "Control Room." The aspects of the auxiliary shutdown panel related to fire protection and the review for conformance to 10 CFR 50, Appendix R (safe shutdown analysis), are included in Section 9.5 of this report.

7.1.5 TMI Action Plan Items

Guidance on implementation of the TMI Action Plan was provided to applicants in NUREG-0737. The items related to instrumentation and control systems are listed below. The specific section of the report addressing each item is indicated in parentheses.

(1) II.D.3 - Direct Indication of PORV and Safety Valve Position (7.5.2.3)

(2) II.E.1.2 - Auxiliary Feedwater System Automatic Initiation and Flow Indication (7.3.3.1)

(3) II.F.1 - Accident Monitoring Instrumentation Positions (4), (5), and (6) (7.5.2.4)

(4) II.F.3 - Instrumentation for Monitoring Accident Conditions (7.5.2.6)

(5) II.K.3.9 - Proportional Integral Derivative Controller Modification (7.7.2.4)

(6) II.K.3.12 - Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (7.2.2.5)

7.2 Reactor Trip System

7.2.1 Description

The reactor trip system (RTS) is designed to automatically limit reactor operation within the limits established in the safety analysis. This function is accomplished by tripping the reactor whenever predetermined safety limits are approached or reached. The RTS monitors variables that are directly related to system limitations or calculated from process variables. Whenever a variable exceeds a setpoint, the reactor is tripped by the insertion of control rods. The RTS initiates a turbine trip when a reactor trip occurs. The RTS consists of sensors and analog and digital circuitry arranged in coincidence logic for monitoring plant parameters. Signals from these channels are used in redundant logic trains. Each of the two trains opens a separate and independent reactor trip breaker. During normal power operation, a dc undervoltage coil in each reactor trip breaker holds the breaker closed. For a reactor trip, the removal of power to the undervoltage coils opens the breakers. Opening either of two series-connected breakers interrupts the power from the rod-drive motor generator sets, and the control rods fall by gravity into the core. The rods cannot be withdrawn until the trip breakers are manually reset, and the trip breakers cannot be manually reset until the abnormal condition that initiated the trip is corrected. Bypass breakers are provided to permit the testing of the primary breakers.

In addition to the automatic trip of the reactor described above, there is also provision for manual trip by the operator. The manual trip consists of two switches. Actuation of either switch removes power from the undervoltage coils and energizes the shunt trip coils of both reactor trip breakers. The shunt trip coils are a diverse means for tripping the reactor trip breakers. The reactor will also be tripped by actuating either of the two manual switches for safety injection.

The generic implications of the Salem anticipated transient without scram (ATWS) events are discussed in Section 7.2.2.4 of this report.

The reactor trips listed below are provided in the Millstone design. The numbers in parentheses after each trip function indicate the coincident logic; for example, two out of three (2/3).

(1) nuclear overpower trips

(a) power range high neutron flux trip (2/4)

(b) intermediate range high neutron flux trip (1/2)

(c) source range high neutron flux trip (1/2)

(d) power range high positive neutron flux rate trip (2/4)

(e) power range high negative neutron flux rate trip (2/4)

(2) core thermal overpower trips

(a) overtemperature ΔT trip (2/4)

(b) overpower ΔT trip (2/4)

(3) reactor coolant system pressurizer pressure and water level trips

(a) pressurizer low pressure trip (2/4)

(b) pressurizer high pressure trip (2/4)

(c) pressurizer high water level trip (2/3)

(4) reactor coolant system low flow trips

(a) low reactor coolant flow (2/3 per loop) (2/4)

(b) reactor coolant pump underspeed trip in any two loops (2/4)

(5) steam generator low-low level trip (2/4)

(6) turbine trip (anticipatory)

(a) low auto stop oil pressure (2/3)

(b) turbine stop valves closed (4/4)

(7) safety injection logic trip (1/2)

(8) manual trip (1/2)

(9) general warning alarm (2/2)

The power range high neutron flux trip has two bistables to initiate reactor trip at separate high flux setpoints. The higher setting trip is active during all modes of operation. The low setting trip provides protection during reactor startup and shutdown when the reactor is below 10% power. The lower setting trip can be manually blocked above 10% power (P-10) and is automatically reinstated when power is reduced below the P-10 interlock setpoint.

The intermediate range trip provides protection during reactor startup and shutdown. This trip can be manually blocked above 10% power (P-10) and is autom-tically reinstated when power is reduced below the P-10 interlock setpoint.

The source range trip provides protection during reactor startup and shutdown when the neutron flux channel is below the P-6 interlock setpoint (6×10^{-11} amp). This trip can be manually blocked above P-6 interlock setpoint and automatically reinstated when power is reduced below P-6 interlock setpoint.

A power range high positive neutron flux rate trip occurs when an abnormal increase in the rate of nuclear power is detected. This trip provides departure from nucleate boiling (DNB) protection against low-worth rod ejection accidents from midpower and is active during all modes of operation.

A power range high negative neutron flux rate trip occurs when an abnormal decrease in the rate of nuclear power is detected. This trip provides protection against two or more dropped rods and is active during all modes of operation.

The overtemperature ΔT trip protects the core against a low departure from nucleate boiling ratio (DNBR). The setpoint for this trip is continuously calculated by analog circuits to compensate for the effects of temperature, pressure, and axial neutron flux difference on DNBR limits.

The overpower T trip protects against excessive power (fuel rod rating protection). The setpoint for this trip is continuously calculated by analog circuits to compensate for the effects of temperature and axial neutron flux difference.

The pressurizer low pressure trip is used to protect against low pressure that could lead to DNB. The reactor is tripped when the pressurizer pressure (compensated for rate of change) falls below a preset limit. This trip may be manually blocked below approximately 10% power (P-7 interlock) to allow startup and controlled shutdown. It is automatically reinstated when power is increased above 10% power.

The pressurizer high pressure trip is used to protect the reactor coolant system against system overpressure. The reactor is tripped when pressurizer pressure exceeds a preset limit.

The pressurizer high water level trip is provided as a backup to the pressurizer high pressure trip and serves to prevent water relief through the pressurizer safety valves. This trip is automatically blocked below approximately 10% of full power (P-7 interlock) to allow startup.

The low reactor coolant flow trip protects the core against DNB resulting from a loss of primary coolant flow. Above the P-7 setpoint (approximately 10% power), a reactor trip will occur if any two loops have low flow. Above the P-8 setpoint (approximately 48% power), a trip will occur if any one loop has low flow. These trips are automatically blocked below the respective interlock setpoints.

The reactor coolant pump underspeed trip protects the reactor core from DNB due to low primary coolant flow. The RCP underspeed trip replaces the undervoltage and underfrequency reactor trips used in some Westinghouse plants. The principle reason for this change is to improve plant availability during voltage dip transients. There is one speed detector mounted on each reactor coolant pump. This trip is automatically blocked below P-7 to permit plant startup.

The steam generator low-low water level trip protects the reactor from loss of heat sink.

A reactor trip on a turbine trip is actuated by two-out-of-three trip fluid pressure signals or by all (four out of four) closed signals from the turbine steam stop valves. A turbine trip causes a reactor trip above 50% power (P-9 interlock). Below 50% power this trip is automatically blocked.

A safety injection signal initiates a reactor trip. This trip protects the core against a loss of reactor coolant or overcooling.

The manual trip is initiated by operation of either of two switches. Each switch de-energizes the undervoltage coils in each reactor trip breaker and shunt coils in these breakers are energized at the same time, which provides a diverse means to ensure that the trip breakers are tripped. Bypass breakers which are closed only when testing the reactor trip breakers are also tripped via their undervoltage and shunt trip coils by a manual reactor trip.

A general warning alarm in both solid-state protection system trains initiates a reactor trip. The general warning alarm is provided for each train of the solid-state protection system and is activated when the corresponding train is being tested or is otherwise inoperable. The trip resulting from the general warning alarm in both trains provides protection for conditions under which both trains of the protection system may be inoperable.

The analog portion of the RTS consists of a portion of the process instrumentation system (PIS) and the nuclear instrumentation system (NIS). The PIS includes those sensors that measure temperature, pressure, fluid flow, and level. The PIS also includes the power supplies, signal conditioning, and bistables that provide initiation of protective functions. The NIS includes the neutron flux monitoring instruments, including power supplies, signal conditioning, and bistables that provide initiation of protective functions.

The digital portion of the RTS consists of the solid-state logic protection system (SSLPS). The SSLPS takes binary inputs (voltage/no voltage) from the PIS and NIS channels corresponding to normal/trip conditions for plant parameters. The SSLPS uses these signals in the required logic combinations and generates trip signals (no voltage) to the undervoltage coils of the reactor trip circuit breakers. The system also provides annunciator, status light, and computer input signals that indicate the condition of the bistable output signals, partial and full trip conditions, and the status of various blocking, permissive, and actuation functions. In addition, the SSLPS includes the logic circuits for testing.

Analog signals derived from protection channels used for nonprotective functions such as control, remote process indication, and computer monitoring are provided by isolation amplifiers located in the protective system cabinets. The isolation amplifiers are designed so that a short circuit, open circuit, or the application of credible fault voltages from within the cabinets on the isolated output portions of the circuit (nonprotective side) will not affect the input signal. The signals obtained from the isolation amplifiers are not returned to the protective system cabinets.

7.2.2 Specific Findings

7.2.2.1 Cable Separation in NSSS Process Cabinets

The staff requested that cable separation inside NSSS cabinets be addressed in the FSAR. The applicant indicated that the FSAR Section 7.2 will be revised to include a reference to WCAP-8872A and confirm that the balance of the plant control systems comply with the NSSS interface criteria. This is a confirmatory item.

7.2.2.2 Trip Setpoint and Margins

The setpoints for the various functions in the reactor trip system are determined on the basis of the accident analysis requirements. As such, during any anticipated operational occurrence or accident, the reactor trip maintains system parameters with the following limits.

(1) minimum departure from nucleate boiling ratio of 1.30.

(2) maximum system pressure of 2750 psi (absolute).

(3) fuel rod maximum linear power of 18.0 kW per foot.

The staff requested detailed information on the methodology used to establish the technical specification trip setpoints and allowable values for the Reactor Protection System (including Reactor Trip and Engineered Safety Feature channels) assumed to operate in the FSAR accident and transient analyses. This includes the following information:

(1) The trip setpoint and allowable value for the Technical Specifications.

(2) The safety limits necessary to protect the integrity of the physical barriers which guard against uncontrolled release of radioactivity.

(3) The values assigned to each component of the combined channel error allowance (e.g., modeling uncertainties, analytical uncertainties, transient overshoot, response time, trip unit setting accuracy, test equipment accuracy, primary element accuracy, sensor drift, nominal and harsh environmental allowances, trip unit drift), the basis for these values, and the method used to sum the individual errors. Where zero is assumed for an error a justification that the error is negligible should be provided.

(4) The margin (i.e., the difference between the safety limit and the setpoint less the combined channel error allowance).

The detailed trip setpoint review will be performed as part of the staff's review of the plant Technical Specifications and will be completed before the operating license is issued. The applicant was requested to provide an evaluation and/or an analysis of the effect of post-accident environmental conditions on the reactor trip system instrumentation (Technical Specification Table 2.2-1) and the engineered safety feature actuation system instrumentation (Technical Specification Table 2.2.2) and its impact on establishing setpoints. This information will be provided for our review with the applicants proposed Technical Specifications.

7.2.2.3 Response-Time Testing

To ensure that the response time of each protective function of the reactor trip system and the engineered safety features actuation system (ESFAS) is within the time limit assumed in the accident analyses, the Technical Specification requires response-time testing at specified intervals. This aspect will be reviewed when the proposed Technical Specifications are available.

7.2.2.4 Design modification for automatic reactor trip using shunt coil trip attachment.

The Westinghouse Owners Group (WOG) has submitted a generic design modification to provide automatic reactor trip system (RTS) actuation of the breaker shunt trip attachments in response to Salem ATWS events. The staff has reviewed and accepted the generic design modification and has identified additional information required on a plant specific basis. The applicant has not however, provided a response to Generic Letter 83-28 which established the requirements for this modification. The resolution of this matter will be addressed in a supplement to this report. This is an open item

7.2.2.5 NUREG-0737 Items II.K.3.12 Confirm Existence of Anticipatory Reactor Trip on Turbine Trip.

The Millstone design includes an anticipatc·y reactor trip on a turbine trip above 50 percent of rated thermal power (P-9 interlock). The staff finds that the design is in compliance with the Action Plan guidelines.

7.2.2.6 Reactor coolant pump (RCP) underspeed trip

The Millstone 3 design utilizes the reactor coolant pump underspeed trip to protect the reactor core from DNB in the event of loss of flow in more than one loop. Since this is a first-of-a-kind parameter used for Reactor Trip System, and each pump only uses one speed sensor, the staff requests that an analysis should be provided to address the conformance with the requirements of IEEE Std. 279. The FSAR should be updated to reflect the deletion of P-17 interlock and the Reactor Coolant pump shaft low low speed trip. This is a confirmatory item.

7.2.2.7 Conformance with Branch Technical Position ICSB-26

Branch Technical Position ICSB-26, "Requirements for reactor protection system anticipatory trip", applies to the entire reactor protection system (RPS) from the sensors to the final actuated device. For sensors located in nonseismic areas the installation (including circuit routing) and design should be such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the RPS and degrade the RPS performance or reliability. There are three groups of RPS related cables which are routing in the turbine building:

1. Turbine trip cause reactor trip input cables
2. Reactor trip to trip the turbine output cables
3. Turbine first stage pressure input to RPS interlock circuits.

The staff requested the applicant to demonstrate that his design is in conformance with BTP ICSB-26 or that exceptions are suitably justified. This is an open item.

7.2.3 Evaluation Conclusion

Later.

7.3 Engineered Safety Features Systems

7.3.1 Engineered Safety Features Actuation System (ESFAS)

The ESFAS is a portion of the plant protection system that monitors selected plant parameters and, on detection of out-of-limit conditions of these parameters, will initiate actuation of appropriate engineered safety features (ESF) systems and essential auxiliary support systems (EAS) equipment. The ESFAS includes both automatic and manual initiation of these systems. Also included with the ESF systems are the control systems that regulate operation of ESF systems following their initiation by the protection system.

The ESFAS is a functionally defined system and consists of:

(1) process instrumentation and control

(2) solid-state and relay logic

(3) ESF test circuits

(4) manual actuation circuits

(5) emergency generator load sequence control logic

The ESFAS includes two distinct portions of circuitry: (1) an analog portion consisting of three to four redundant channels per parameter or variable to monitor various plant parameters such as reactor coolant and steam system pressures, temperatures, and flows and containment pressure and (2) a digital portion consisting of redundant logic trains that receive inputs from the analog protection channels and perform the logic to actuate the ESF equipments. The ESFAS is composed of the NSSS circuits designed by Westinghouse and the BOP circuits designed by Stone and Webster Engineering Corporation.

The actuation signals for each of the ESFAS functions are listed below. The numbers in parentheses after each actuation channel indicate the coincident logic; for example, two out of four (2/4).

(1) Safety Injection

- (a) Manual (1/2)
- (b) High-1 Containment Pressure (2/3)
- (c) Low Compensated Steam Line Pressure (2/3 in any line)
- (d) Low Pressurizer Pressure (2/4)

(2) Containment Depressurization

- (a) Manual (2/4)
- (b) High-3 Containment Pressure (2/4)

(3) Containment Isolation

(a) Phase A Isolation

Safety Injection (same as item (1) above) manual (1/2)

(b) Phase B Isolation

High-3 Containment Pressure (2/4)

Manual (2/4)

(4) Steam Line Isolation

(a) Low Compensated Steam Line Pressure (2/3 in any line)

(b) High-2 Containment Pressure (2/3)

(c) High Negative Steam Pressure Rate (2/3 in any line)

(d) Manual (1/2 for all lines or 1/1 for each valve)

(5) Feedwater Line Isolation

(a) Safety Injection (same as item (1) above)

(b) High steam generator level (2/3 in any generator)

(c) Low Tavg (2/4) coincident with reactor trip

(6) Auxiliary Feedwater System Actuation

The motor driven auxiliary feedwater pumps will be started on any of the following signals:

(a) Safety injection (same as Item 1 above)

(b) Low-Low steam generator level (2/4) in any generator

(c) Loss of power (2/4) undervoltage at 4.16 KV buses

(d) Manual actuation (1/1)

The turbine-driven auxiliary feedwater pump will be started on any of the following signals:

(a) Low-low level (2/4) in two steam generators

(b) Loss of power (2/4) undervoltage at 4.16 KV buses

(c) Manual actuation (1/1)

(7) Control Building Isolation

(a) High-High radiation in air intake (1/2)

(b) High-1 containment pressure (2/3)

(c) High chlorine in air intake (1/2)

(d) Manual safety injection (1/2)

(e) Manual actuation (1/2)

7.3.2 ESF and EAS System Operation

The following systems are provided.

A. Engineered Safety Features Systems

1. Emergency core cooling system (ECCS)

2. Containment depressurization system

   a. Quench spray system

   b. Containment recirculation system

3. Containment isolation system including main steam and feedwater isolation

4. DBA hydrogen recombiner system

5. Supplementary leak collection and release system

6. Auxiliary feedwater system

7. ESF filtration system

   a. Control room ventilation system

   b. Fuel building exhaust system

   c. ESF equipment areas ventilation and filtration system

B. Essential Auxiliary Support Systems

1. Service water system

2. Reactor component cooling system

3. Emergency onsite power supply system

4. Emergency diesel generator support systems

7.3.2.1 <u>Emergency Core Cooling System</u>

The emergency core cooling system (ECCS) cools the reactor core and provides shutdown capability for pipe breaks in the reactor coolant system (RCS) that cause a loss of primary coolant greater than that which can be made up by the normal makeup system, for rod cluster control assembly ejection, for pipe breaks in the secondary coolant system and for steam generator tube failure. The primary function of the ECCS is to remove the stored and fission product decay heat from the reactor core during accident conditions. The ECCS consists of the centrifugal charging, safety injection and residual heat removal pumps, accumulators, containment recirculation pumps, refueling water storage tank (RWST), the associated piping, valves, and instrumentation.

The ECCS provides reactor shutdown capability for the accidents described above by injecting borated water into the RCS. The system's safety function can be performed with a single active failure (short term) or passive failure (long term). The emergency diesel generators supply power if offsite power is unavailable.

The safety injection signal will start the diesel generators and automatically initiate the following actions in the ECCS:

(1) starts charging pumps

(2) opens RWST suction valves to charging pumps

(3) opens charging pumps to RCS cold leg injection headers isolation valves

(4) closes normal charging path valves

(5) closes charging pump miniflow valves

(6) starts safety injection pumps

(7) starts residual heat removal (RHR) pumps

(8) opens any closed accumulator isolation valves

(9) closes volume control tank outlet isolation valves

Switchover from the injection mode to recirculation mode involves the procedures described below. The changeover from the injection mode to recirculation mode is initiated manually by operator action from the main control room. The switchover procedures are outlined as follows:

(1) From Injection to cold leg recirculation

(a) The RHR pumps are stopped automatically when RWST level reaches low-low setpoint.

(b) Align valves associated with RHR pumps and containment recirculation pumps for cold leg recirculation mode.

(c) close the safety injection pump miniflow valves

(d) Align safety injection and charging pumps to the containment recirculation pump discharge

(e) Isolate the refueling water storage tank.

(2) After approximately 15 hours, cold leg recirculation is terminated and hot leg recirculation is initated.

(a) Align containment recirculation pump to deliver directly to the RCS through the hot leg injection heater.

(b) Align containment recirculation pump to deliver to the RCS via the safety injection pumps.

7.3.2.2 Containment Depressurization System

The containment depressurization system consists of the quench spray system and the containment recirculation spray system. Subsequent to a design basis accident (DBA), the quench spray pumps are started automatically on receipt of a containment depressurization actuation (CDA) signal. The isolation valves in the quench spray discharge heaters and the chemical addition tank open on receipt of a CDA signal. Each redundant quench spray subsystem draws water independently from the RWST. Sodium hydroxide solution is added to the quench spray by direct gravity feed from the chemical addition tank. The quench spray pumps are stopped automatically on receipt of a RWST Low-3 signal. Before the RWST reaches Low-3 level, the RWST Low-2 signal alerts the operator to take manual action for changeover from injection mode to recirculation mode. The containment recirculation pumps start automatically on a CDA signal after about a 5 minute time delay. The containment recirculation pumps take suction from the containment sump. Two of the four containment recirculation pumps perform the containment spray function to replace the quench spray pumps during the recirculation mode. The other two pumps are used for cold leg injection.

7.3.2.3 Containment Isolation System Including Main Steam And Feedwater Isolation

The safety function of the containment isolation system (CIS) is to automatically isolate the process lines penetrating the containment structure. The CIS is designed to limit the release of radioactive materials from the containment following an accident.

The CIS is automatically actuated by signals developed by the ESFAS in two phases; Phase A containment isolation and phase B containment isolation. Phase A isolates all nonessential process lines penetrating the containment. Phase B isolates all other process lines not included in phase A containment isolation, except for the safety injection and containment spray lines.

Containment isolation valves, which are equipped with power operators and are automatically actuated, may also be controlled individually by manual switches in the control room. Containment isolation valves with power operators are provided with an open/closed indication, which is displayed in the control room at the main control board and the safeguard status panel. All electric power supplies and equipment necessary for containment isolation are Class 1E.

The main steam line isolation signal is generated on low steam line pressure, high-2 containment pressure or high negative steam pressure rate. A manual bypass permissive is provided for the low steam line pressure signal for use during normal plant cooldowns and heatups. The high negative steam line pressure rate is used to initiate main steam isolation when the low steam line pressure signals are bypassed during normal plant start up and shutdown. The main steam isolation trip valves are Y-pattern type globe valves designed to prevent main steam flow in both the forward and reverse directions. Closing forces are provided by steam pressure from the main steam line. Each main steam isolation valve is closed by redundant logic trip signals. The main steam isolation valves are capable of being tested on-line by partial closure of the valve.

Feedwater line isolation is provided to terminate main feedwater following a pipe rupture or excessive feedwater flow event. The feedwater line isolation signal is generated on safety injection, high steam generator water level, or low reactor coolant temperature coincident with reactor trip. Upon receipt of this signal, the main feedwater isolation valves and other valves associated with the main feedwater lines are closed. Redundant actuation systems are provided for each valve operator and receive closure signals from the two redundant ESFAS logic trains. However, the applicant stated that the feedwater isolation valve schematic has not been finalized. The staff will review this design later.

7.3.2.4 DBA Hydrogen Recombiner System

The DBA hydrogen recombiner system controls the building of hydrogen gas inside the containment. The DBA hydrogen recombiner system consists of hydrogen monitors and hydrogen recombiners. The applicant has not completed the design on this system. The staff will review this design later.

7.3 2.5 Supplementary Leak Collection and Release System (SLCRS)

Millstone 3 uses a dual containment design. There is a containment enclosure building outside the containment. The SLCRS is designed to maintain the containment enclosure building at negative pressure of 0.25 inch wg after a design basis accident (DBA). The SLCRS also maintains part of contiguous buildings - main steam valves building, engineering safety features building, hydrogen recombiner building, and auxiliary building under a negative pressure following a design basis accident. The SLCRS exhausts air from these areas, filters and removes particulate and gaseous iodine from the air before discharge to the atmosphere. The safety injection signal opens the SLCRS train A filter bank supply and discharge dampers and starts the train A exhaust fan. Train B serves as a backup train. Train B will start automatically if train A air flow is below a preset limit. High differential pressure across the train A filter bank is alarmed in the control room. High radiation signal at operating train filter outlet will alert the operator to direct the exhaust flow through the standby filter bank.

7.3.2.6 Auxiliary Feedwater System

The function of the auxiliary feedwater system (AFWS) is to provide an adequate supply of water to the steam generators if the main feedwater system is not available. The AFWS consists of two-motor-driven pumps and one turbine-driven pump with associated valves, controls, and instrumentation. Each motor-driven pump supplies water to two of the four steam generators while the turbine-driven pump supplies water to all four steam generators. The auxiliary feedwater (AFW) actuation system will automatically start the pumps and provide feedwater to the steam generators. The initiating conditions are listed in Section 7.3.1, item (6). The AFW pump suction is normally supplied from the seismic Category I demineralized water storage tank. An additional source of water is available from a non-seismic Category I condensate storage tank. The service water is the long term safety grade sour^e of auxiliary feedwater which can be manually connected by spool pieces.

The AFWS can be manually initiated and controlled from the main control board or the auxiliary shutdown panel. The AFWS control is addressed in Section 7.4 of this report.

The amount of flow to any steam generator is limited by cavitating venturis located in the auxiliary feedwater line to each steam generator. The cavitating venturis will prevent runout flow to a depressurized steam generator. Manual isolation of AFW flow to a depressurized steam generator can be performed from the main control board or the auxiliary shutdown panel.

7.3.2.7 <u>ESF Filtration System</u>

1. <u>Control Room Ventilation System</u>

   The control room ventilation system includes the control room air conditioning system, the instrument room and computer room air conditioning system, control room emergency ventilation and pressurization system, and other control building ventilation systems. The control room is normally maintained at a slightly positive pressure. The pressure is maintained by redundant isolation valves or dampers on all inlet and exhaust openings. Redundant radiation monitors and chlorine gas detectors are located at the control room air intake. High radiation or high chlorine will automatically isolate the control room. After isolation, compressed air from air storage tanks is used to maintain a positive air pressure during first hour. After an hour, outdoor air is introduced to the control room through redundant emergency filtration trains. The control room air intake is also provided with smoke detectors to actuate smoke alarms. Smoke can be purged by the purge ventilation system.

2. <u>Fuel Building Exhaust System</u>

The fuel building filter banks are normally bypassed by the unfiltered exhaust fan. During refueling and in the event of high radiation, the fuel building exhaust is manually diverted to the fuel building filter bank. Either train A or train B is operated with the other train at standby.

3. Equipment Areas Ventilation and Filtration System

The ESF equipment areas ventilation and filtration system controls and minimizes the potential for spread of airborne radioactive material within the building. On receipt of a safety injection signal (SIS) or containment depressurization signal (CDS), all the nonsafety related ventilation systems will be shutdown and isolated except the areas served by the ESF filtration system. These areas include the charging pump rooms, component cooling water pump rooms, safety related heat exchangers areas, rod control areas and safety related motor control center areas. The ESF filtration system includes redundant trains. Each train consists of exhaust fans, filter banks and the associated ductwork and dampers. Each train is powered from a separate emergency bus. The exhaust air can be directed through the auxiliary building filters to atmosphere. The filter inlet dampers from the charging pump and component cooling pump areas are in parallel and fail open on loss of power or instrument air. The filter inlet dampers from other safety-related areas are in series and fail closed on loss of power or instrument air. The ESF filter banks can be manually controlled from the control room or at the switchgear. Control transfer switches are provided at the switchgear. An alarm is sounded when LOCAL control is selected. High differential pressure across a filter bank is alarmed in the control room.

7.3.2.8 Service Water System

The service water system performs both safety and non-safety functions by providing cooling water for heat removal components during all modes of operation. The service water system consists of two trains. Each train contains two half capacity service water pumps, two strainers, two booster pumps and associated piping and valves. One pump in each train is operated with the other in standby. The service water system is designed to meet the single failure criterion. Power is supplied to redundant pumps from separate emergency buses. On receipt of safety injection signal or loss of power signal, the water supply lines to the non-safety related equipment are isolated. On receipt of containment depressurization actuation signal, the water supply lines to the reactor plant component cooling water heat exchangers are isolated and the water supply lines to the containment recirculation coolers are opened.

7.3.2.9 Reactor Components Cooling Systems

The cooling systems for reactor components consist of the charging pumps cooling system, safety injection pumps cooling system, reactor plant component cooling water (RPCCW) system and other nonsafety related component cooling systems. These systems are used individually or in combination to provide cooling water for heat removal from reactor plant components.

The charging pumps cooling system is a safety-related closed loop cooling system which transfers the heat load from the charging pumps lubricating oil coolers to the service water system. This system consists of two full capacity pumps, two coolers, a surge tank, and associated piping and valves. On failure of the operating cooling pump the standby pump will automatically start. Either pump can supply cooling water to any charging pump oil cooler.

The safety injection pumps cooling system is a safety-related closed loop cooling system which cools the safety injection pumps bearing oil. This system consists of two full capacity pumps, two coolers, a surge tank and associated piping and valves.

The reactor plant component cooling water (RPCCW) system is a closed loop cooling system. The RPCCW system includes three half capacity pumps and heat exchangers, a surge tank, a chemical addition tank, and associated piping and valves. Two redundant trains serve those components essential for safe shutdown. One pump and heat exchanger are provided as a spare. The pump can be manually connected to either train's emergency bus. The spare pump motor breaker has to be racked out from one train cubicle and then racked into the other train cubicle to prevent~~s~~ a cross tie between redundant buses. An electrical interlock prevents simultaneous operation of two pumps on the same train. Redundant pressure switches are located at the nonsafety portion water supply header to detect a drop in pressure which indicates a rupture of nonsafety-related system piping. Low pressure automatically isolates component water to the nonsafety portions of the system.

7.3.2.10 Emergency Onsite Power Supply System

The emergency onsite power supply system consists of two 4.16 KV diesel generators, two 4.16 KV ESF buses, various ESF and non-ESF 480V buses, motor control centers and 208/120V power panels. There are four 120 VAC safety related power distribution panels for safety related vital instrumentation and control loads. Each power panel has a separate rectifier/inverter. The DC power system consists of four Class 1E DC power panels (2 panels per train) and two non-safety DC power panels. Each Class 1E DC power panel consists of a battery bank and a static battery charger. One spare battery charger per train is available to replace either of the two chargers in that train.

7.3.2.11 Emergency Diesel Generator Support Systems

The diesel generator fuel oil system, the diesel engine cooling water system, the diesel generator starting air system, the diesel engine lubrication system, and the diesel generator air intake and exhaust system are essential auxiliary support systems. The evaluation of these systems are addressed in Section 9.5 of this report.

7.3.3 Specific Findings

7.3.3.1 NUREG-0737 Item II.E.1.2, AFWS Automatic Initiation and Flow Indication

The automatic system used to initiate the operation of the auxiliary feedwater system is part of ESFAS. The redundant actuation channels that provide signals to the pumps and valves are physically separated and electrically independent. Redundant trains are powered from independent Class 1E power sources. The initiation signals and circuits are testable during power operation, and the test requirements are included in the plant Technical Specifications. Manual initiation and control can be performed from the main control board or the auxiliary shutdown panel. No single failure within the manual or automatic initiation system for the auxiliary feedwater system will prevent initiation of the system by manual or automatic means. The environmental qualification is addressed in Section 3.11 of this report.

Redundant auxiliary feedwater flow instrument channels are provided for each steam generator. Each channel is powered from a separate Class 1E power source. Auxiliary feedwater flow indicators are located at the main control board and the auxiliary shutdown panel. The staff concludes that the design satisfies the requirements of NUREG-0737, item II.E.1.2.

7.3.3.2 Test of Engineered Safeguards P-4 Interlock

On November 7, 1979, Westinghouse notified the Commission of an undetectable failure that could exist in the engineered safeguards P-4 interlocks. Test procedures were developed to detect failures that might occur. The procedures require the use of voltage measurements at the terminal blocks of the reactor trip breaker cabinets.

The staff raised a concern on the possibility of accidental shorting or grounding of safety system circuits during testing of the P-4 interlocks. The applicant has committed to incorporate built-in test features to facilitate testing of the P-4 interlock. This is a confirmatory item.

7.3.3.3 Level Measurement Errors Resulting From Environmental Temperature Effects on Level Instrument Reference Legs

The staff requested that the applicant evaluate the effects of high temperatures in reference legs of water level measurement systems due to high energy-line breaks. This issue was addressed for operating reactor through IE Bulletin 79-21. In FSAR Amendment No. 5, the applicant committed to insulate the steam generator reference legs in response to the heat up concern addressed in IE Bulletin 79-21. The staff finds this acceptable.

7.3.3.4 Steam Generator Level Control and Protection

Three steam generator level channels are used in a two-out-of-three logic for isolation of feedwater on high steam generator level. One of the three level channels is used for control. This design for actuation of feedwater isolation does not meet the requirements of Paragraph 4.7 of IEEE 279 on "Control and Protection System Interaction" in that the failure of the level channel used for control could require protective action and the remainder of the protection system channels would not satisfy the single-failure criterion. The applicant has not responded to this concern. This is an open item.

7.3.3.5 IE Bulletin 80-06 Concerns

As was done for operating reactors through IE Bulletin 80-06, the staff requested that the applicant review all safety systems to determine if any safety equipment would change state after reset. In FSAR Amendment No. 5, the applicant stated that the requested reviews have been performed and that safety-related equipment will remain in its associated emergency mode following reset. The conclusions of the applicant review are as follows:

1. All equipment receiving an ESF actuation signal directly and not via the emergency diesel sequencer will remain in the emergency mode. After the equipment receives an ESF signal, it is driven to its emergency position. The ESF signal can be reset, and the equipment will remain in the emergency mode.

2. To change the equipment from its emergency position, the ESF signal must be reset and the equipment control switch must be operated.

3. All equipment receiving a loss of offsite power (LOP) actuation signal via the sequencer will go to its emergency position and remain there as in 1. and 2. above, except the quench spray and recirculation spray pump motors. The reason for this is that the SIS signal cannot be reset until after a time delay which assures that load sequenced by a SIS signal will have started, however the CDA signal can be reset at any time. If the CDA signal is reset before the quench spray and recirculation spray pumps are actuated by the sequencer after a LOP, then the quench spray and recirculation spray pumps will not start. If after a CDA signal is received and the quench spray or recirculation spray pump motors start, then

resetting the CDS signal will not stop the motors. The pumps motors can be stopped with their control switch if the CDA signal is not present. If the CDA output signal is reset and blocked before the pumps motors are actuated, then this is treated as a bypassed or inoperable status and annunciated as part of the Regulatory Guide 1.47 alarms.

The staff finds that the design is consistent with the intent of the bulletin. The bulletin requires a confirmatory test to verify the conclusions of this review. This is a confirmatory item subject to the applicant's commitment to perform a confirmatory test.

7.3.3.6 Containment Isolation For The Main Steam Lines To The Turbine Of The AFW Pump

General Design Criteria 57 requires that each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. The main steam lines to the turbine of the AFW pump have a motor operated check stop valve in parallel with an air-operated bypass valve, both of which are remote manually operated. The staff is concerned that the bypass valves (AOV84A, B, & D) are not supplied power from a Class 1E power source. Therefore, isolation of the bypass valves cannot be assured. This is an open item.

7.3.3.7 Letdown Line Relief Valve

The staff raised a concern that the relief valve located on the letdown line would relieve primary coolant to the reactor drain tank in the event the isolation valve inside containment did not close on a containment isolation signal or if the outside containment isolation valve failed closed. The applicant has not responded to this concern. This is an open item.

7.3.3.8 Control Building Isolation Reset

The staff raised a concern that whenever an automatic isolation signal is bypassed, all other control building isolation signals are blocked because a single relay is used for all isolation functions. The applicant proposed a design modification which provides individual devices for each isolation area. The staff finds the proposed design modification acceptable. This is a confirmatory item subject to documentation of these changes.

7.3.3.9 Power Lockout Feature For Certain Motor Operated Valves

Certain valve operators can be power lockout during plant operation from the rear panel of the main control board. The staff raised a concern on the power lockout feature that in this mode the contacts for the motor operated valve would pick up and seal in if the operator attempted to open the valve. Since this condition is not detectable, a single failure in the power lockout circuit could lead to inadvertent closure or opening of the valve. The applicant proposed a modified design which adds a auxiliary contact to de-energize the control circuit. The staff finds the proposed modification acceptable. This is a confirmatory item subject to documentation of the drawing changes.

The following motor operated valves will be affected:

VALVE NO.	FUNCTION	DRAWING NO.
3SIH*MV 8806	SI Pumps Suction/RWST	ESK-GMF
3SIL*MV 8840	RHR Pumps/Hot Leg	ESK-GNM
3SIH*MV 8802A	SI Pump Disch/Hot Leg	ESK-GMR
3SIH*MV 8802B	SI Pump Disch/Hot Leg	ESK-GMS
3SIH*MV 8835	SI Pumps Disch/Cold Leg	ESK-GML
3SIL*MV 8809A	RHR Pump Disch/Cold Leg	ESK-GME
3SIL*MV 8809B	RHR Pump Disch/Cold Leg	ESK-GNA
3SIH*MV 8813	SI Pumps Recir/RWST	ESK-GMN
3RHS*MV 8716A	RHR Pump Disch Cross over/Hot/Cold Leg	ESK-GNJ
3RHS*MV 8716B	RHR Pump Disch Cross over/Hot/Cold Leg	ESK-GNK
3SIH*MV 8821A	SI Pump Disch Cross over/Hot/Cold Leg	ESK-GMJ
3SIH*MV 8821B	SI Pump Disch Cross over/Hot/Cold Leg	ESK-GMK

7.3.3.10 Failure Modes And Effects Analyses Of ESFAS

The applicant referred the Westinghouse topical report WCAP-8584, Failure mode and effects analysis (FMEA) of the engineered safety feature actuation system, for ESF systems equipment (FMEA) within the NSSS scope of supply. For balance of plant equipment, a fault tree analyses, based on actual wiring diagrams and components of the plant, were performed. The applicant concluded that the single failure criterion of IEEE Standard 279 requirements was met for the Class 1E instrumentation and control portions of the safety related systems.

Because the FMEA for the NSSS was performed using assumptions on the BOP design, the staff requested the applicant to review that the interfa e requirements of Appendix B and C of WCAP-8584 are met. The applicant confirmed that the BOP design complies with the interface requirements of Appendix B and C of WCAP-3584. This is a confirmatory item subject to documentation in the FSAR.

7.3.3.11 Non-Class 1E Control Signals To Class 1E Control Circuits

The staff requested the applicant to provide a list of non-Class 1E control signals that are used as inputs to Class 1E control circuits and justification that these non-Class 1E signals are either bypassed by the ESF actuation signal, or that the non-Class 1E signal can only act to the safe direction and therefore would not degrade safety systems. This is an open item.

7.3.3.12 Isolators Used In The BOP Design For Isolation Between Safety And Non-Safety Related Systems

Millstone 3 utilizes multiplexers for information processing. Portions of the radiation monitoring system are safety related and use safety-related microprocessor which has interface with the nonsafety-related radiation monitorjng computer via qualified isolators. The staff requested additional information on the qualification of the isolators used for the radiation monitoring system. This is an open item.

7.3.3.13 Feedwater Isolation and Control Valves

The staff requested detailed schematic drawings for feedwater isolation valves and feedwater control valves. The applicant stated that detailed drawings will not be available until March 1984. This is an open item.

7.3.3.14 Sequencer Deficiency Report

On August 19, 1983, the Vitro Laboratories, the manufacturer of Millstone emergency power loading sequencer, filed a 10 CFR 21 deficiency report. The report indicated that the design of the auto test circuitry does not permit the proper output in response to LOCA events in some circumstances. Specifically, one reset function was omitted from the input LOCA time delay. The result is that LOCA events occurring during the portion of the auto test cycle will not actuate some output relays. The applicant has also filed a 10 CFR 21 report to note this deficiency. This item is a confirmatory item subject to implementation of the required corrective action.

7.3.3.15 BOP Instrumentation And Control System Testing Capability

The FSAR Sections 7.2.2.2.3 and 7.3.2.2.5 describe the capability for testing the reactor trip system and the engineered safety feature system. Most of the descriptions are based on NSSS scope of supply equipment. It is not clear whether all the BOP instrumentation and control systems satisfy the same criteria. The staff cited an example on the refueling water storage tank (RWST) level measurement which is a BOP design. The low-low loop signal from one-out-of-two level switches will automatically stop the residual heat removal pump. The empty tank signal from one-out-of-two level switches will automatically stop the quench spray pumps. The testing of these actuation logic circuits are not discussed in the FSAR and they are not tested by the same method as NSSS ESF instrument systems. The staff requested that the applicant performs a thorough evaluation on the BOP safety related instrumentation and control systems with respect to testing capabilities, identify any instrument channels which cannot be tested as described in Sections 7.2.2.2.3 and 7.3.2.2.5, and to justify that the design is in conformance with the testing requirements of GDC-21. This is an open item.

7.3.3.16 Spare Component Cooling Water (CCW) Pump

A spare CCW pump is provided to allow continued plant operation when one of the two CCW pumps is out of service. The spare pump can replace either pump in the redundant CCW loops and maintain the required safety train separation of the electrical power supplies and control circuits. The applicant was requested to include the proposed Technical Specifications testing of the Spare pump breaker and the CCW surge tank level instruments. This is a Technical Specification item.

7.3.4 Evaluation Conclusion

Later.

7.4 Systems Required for Safe Shutdown

7.4.1 Description

This section describes the equipment and associated controls and instrumentation of systems required for safe shutdown. It also includes controls and instrumentation located outside the main control room that enable safe shutdown of the plant in the event the main control room is evacuated.

7.4.1.1 Safe Shutdown System

The systems required for safe shutdown are those required to (1) control the reactor coolant system temperature and pressure, (2) borate the reactor coolant, and (3) provide adequate residual heat removal. There are two kinds of shutdown conditions: hot standby and cold shutdown. Hot standby is a stable condition of the plant achieved shortly after a programmed or emergency shutdown of the plant. Cold shutdown is a stable condition of the plant achieved after the residual heat removal process has brought the primary coolant temperature below 200°F. For either case, the following systems are required for achieving and maintaining safe shutdown condition.

1. Emergency Class 1E electrical power supply systems
2. Auxiliary feedwater system
3. Residual heat removal system
4. Boration and reactor coolant inventory control system
5. Reactor coolant pressure relief system
6. Steam generator PORV and bypass valves
7. Component cooling water system
8. Service water system
9. Safety-related heating, ventilation, and air conditioning systems

To achieve and maintain safe shutdown, the reactor and the turbine are tripped. Automatic protection and control system functions are discussed in Sections 7.2 and 7.3. The controls and the indicators for all of the equipment listed above are provided in the main control room. In addition, an auxiliary shutdown panel is provided that allows the plant to be maintained in a hot standby condition or taken to cold shutdown should the main control room become uninhabitable.

The safe shutdown design basis for Millstone 3 is cold shutdown. The plant can be taken from no load temperature and pressure to RHR system initiation within 36 hours following and condition II, III or IV events using only safety grade systems, with or without off-site power, with a single failure and with limited operator action outside the control room. Safe shutdown includes boration and depressurization of the primary coolant system. During the first phase of cooldown, heat removal is accomplished via the steam generator power operated relief valves (PORV) and the auxiliary feedwater system. Boration is accomplished by the charging pumps injecting borated water into the reactor coolant system. Gravity drain lines are connected from the boric acid tanks to the charging pumps header. Control of the boration rate is accomplished by throttling valves in the flow paths from the charging pumps to the high head safety injection lines. A parallel and series arrangement of Class 1E solenoid valves is provided for the reactor vessel head letdown path. Depressurization of the reactor coolant system is accomplished by the solenoid operated pressurizer PORVs. When the reactor coolant system temperature and pressure are reduced to about 350°F and 425 psig, RHR is initiated and cooldown proceeds to the normal plant cold shutdown condition.

7.4.2 Specific Findings

7.4.2.1 Turbine-driven auxiliary feedwater pump control transfer

During our drawing review, the staff raised a concern on turbine-driven auxiliary feedwater pump control transfer design. Whenever the control for the turbine-driven auxiliary feedwater pump is transferred from the main control room to the auxiliary shutdown panel, the turbine-driven feedwater pump starts automatically. The applicant has proposed a design modification for the control circuitry to prevent inadvertent starting the pump during the transfer. The staff finds the modified design acceptable.

7.4.2.2 Auxiliary Feedwater Control

The staff's review on the AFWS included the following considerations:

(1) automatic initiation (discussed in Section 7.3)

(2) capability of controlling flows to establish and maintain steam generator level

(3) capability of controlling the steam generator pressure

(4) capability of isolating a faulted steam generator resulting from feedwater or steam line breaks

(5) capability for post trip control from auxiliary shutdown panel

The auxiliary feedwater flow to each steam generator is through the normally opened control valves. Each control valve can be manually adjusted from the control room as dictated by the steam generator water level and auxiliary feedwater flow rate. The control valves also can be manually adjusted from the auxiliary shutdown panel. The auxiliary feedwater is fed to the steam generators through a connection downstream of the main feedwater stop-check valves. The auxiliary feedwater has sufficient water supply to hold the unit at hot standby for up to 10 hours. The reactor coolant temperature can be reduced to 350°F in 6 hours, at which time the residual heat removal system will be initiated.

During plant cooldown, the main steam power operated relief valves are automatically controlled by steamline pressure. Manual control of the PORVs is provided to control the steam generator pressure to permit cooldown from the main control board or the auxiliary shutdown panel. Auxiliary feedwater flow to the steam generators is limited by flow venturies located in each auxiliary feedwater line. These venturies are sized to restrict the flow to a depressurized steam generator. Two isolation valves are provided in each of the auxiliary feedwater supply lines. One valve is powered by train A power source, the other valve is powered by train B power source. The isolation valves can be operated either from the main control board or the auxiliary shutdown panel.

Indications are provided at the auxiliary shutdown panel for steam generators level and pressure, auxiliary feedwater flow and demineralized water tank level. The capability is provided to control the auxiliary feedwater pumps and to isolate a depressurized loop as well as for post trip control of the auxiliary feedwater system at the auxiliary shutdown panel. Based on our review, we find that the auxiliary feedwater control system design is acceptable.

7.4.2.3 Remote shutdown capability

GDC-19 requires that equipment at appropriate locations outside the control room be provided to achieve a safe shutdown of the reactor. The Standard Review Plan (SRP) Section 7.4 provides guidance on conformance to the GDC-19 requirements. The design should provide redundant safety grade capability to achieve and maintain safe shutdown from a location or locations remote from the control room, assum ng no fire damage to any required systems and equipment and assuming no accident has occurred. The remote shutdown station equipment should be capable of maintaining functional operability under all service condition postulated to occur including the seismic event. The remote shutdown stations and the equipment used to maintain safe shutdown should be designed to accommodate a single failure.

In the FSAR Section 7.4.1.3, the applicant states that the design basis for control room evacuation does not consider a single failure. The staff finds the applicant's design basis for remote shutdown capability unacceptable. The FSAR Table 7.4-1, "Instruments and Controls Outside Control Room For Cold Shutdown" has not identified the transfer switches whether from train A equipment or from train B equipment. The staff requested the applicant to clarify the design criteria for remote shutdown station, and provide detailed layout drawings for transfer switch panels and the auxiliary shutdown panel. The applicant should also address the isolation, separation, qualification, and transfer/override provisions of the remote shutdown station in Section 7.4 of the FSAR. Detailed schematics related to remote shutdown operation should be provided for staff review. This is an open item.

7.4.2.4 Testing for Remote Shutdown Operation

During the review process, a concern was raised by the staff regarding the remote shutdown capability and the need for a test to verify design adequacy. The applicant stated that emergency procedures will be prepared to include remote shutdown, and a test will be conducted during startup testing to confirm the capability for remote shutdown. The test description is outlined in FSAR Table 14.2-2 item 25. The staff finds the applicant's commitment for remote shutdown operation testing acceptable.

7.4.3 Evaluation Conclusion

Later.

7.5 Information Systems Important To Safety

7.5.1 Description

The applicant has conducted an analysis to identify the appropriate variables for the operator to monitor conditions in the reactor coolant system, the secondary heat removal system, the containment system, the engineered safety features systems and the safe shutdown systems. The safety-related display instrumentation system provides the information necessary for the operator to perform the required manual safety functions following a reactor trip. It provides information for all operating conditions, including anticipated operational occurrences and accidents and post accident conditions.

The instrumentation identified in FSAR Table 7.5-1 ~~also~~ includes the following information for each variable identified:

1. Instrument range
2. Environmental qualification ~~status~~
3. Seismic qualification ~~status~~
4. Display methodology
5. Type and category (per the definition in R.G. 1.97 Rev. 2)
6. Schedule for implementation

The ~~detailed~~ qualification ~~status~~ for ~~each~~ these instruments ~~will be~~ is addressed in Section 3.10 and 3.11 of this report.

7.5.2 Specific Findings

7.5.2.1 Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation (IE Bulletin 79-27)

The staff requested that the applicant review the adequacy of emergency operating procedures to be used by control room operators to attain safe shutdown on loss of any Class 1E or non-Class 1E bus~~es~~ supplying power to safety-or nonsafety-related instrument and control systems. This issue was addressed for operating reactors through IE Bulletin 79-27. In FSAR Amendment No. 5, the applicant responded that Millstone Unit 3 can achieve a cold shutdown condition without the use of any non-class 1E power. All the equipment required to achieve a cold shutdown is redundant and is powered from redundant Class 1E buses, which satisfies the single failure criterion. ~~is satisfied,~~ However, the staff pointed out that loss of a single instrument bus could affect the interlock circuits to ~~block~~ isolate both trains of Residual Heat Removal system, therefore, the applicant's response ~~has~~ did not adequately addressed the IE Bulletin 79-27 concerns identified in ~~concerns~~. The staff requested that the applicant ~~should~~ re-evaluate his response ~~approach~~ to resolve this concern. Additional information is required to address items requested in IE Bulletin 79-27. This is an open item.

(or "conformity ..."? however ...)

7.5.2.2 Bypass and inoperable status panel

The FSAR Section 1.8 states that Millstone 3 design is in conformance with R.G. 1.47, Bypassed and inoperable status indication for nuclear power plant safety systems. During the ~~ICSB~~ review ~~meeting,~~ the staff ~~audit~~ reviewed ~~some of~~ design drawings which contain ~~partial~~ information of the bypass and inoperable status panel. However, there is no information in the FSAR to describe the system. The staff requested that the applicant provides the descriptive information in Section 7.5 of the FSAR to demonstrate ~~the~~ conformance with R.G. 1.47. This is an open item.

7.5.2.3 NUREG-0737, Item II.D.3 - Direct Indication of Relief and Safety Valve Positions

The two pressurizer power operated relief valves (PORV) are operated automatically or by remote manual control. Each valve is provided with positive open/closed indication lights in the control room. The three safety valves are also provided with positive open/closed indication lights. The ~~Temperature~~ temperature in each of the safety valve and PORVs discharge lines ~~are~~ is measured and indicated in the control room. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve. High temperature will be alarmed in the control room. The valves position indicating limit switches are seismically and environmentally qualified. The staff finds that the design is in conformance with the action plan guidelines, and is, therefore, ~~is~~ acceptable.

7.5.2.4 NUREG-0737 Item II.F.1 Accident Monitoring Instrumentation Position (4), (5), and (6)

Position (4), (5), and (6) of this action plan item require installation of ~~a~~ the extended range containment pressure monitors, containment water level monitors, and containment hydrogen concentration monitors. Table 7.5-1 of the FSAR indicated that information on these parameters will be provided later. This is an open item.

~~Position (4) requires extended range for containment pressure measurement. Millstone 3 provides two wide range containment pressure measurements. The extended range is 0-200 psia. Instruments are seismic and environmentally qualified. Two indicators and one dual recorder are provided in the control room. However, the~~

I render this is insufficient / What would be acceptable. My guess that any [illegible] would be ok. I don't think it'd make these an open or confirmatory items.

applicant has not ~~defined~~ completed the table to show the accuracy and response time ~~specifications. Additional information is required.~~ The FSAR should be updated to for this instrument.

~~Position (5) requires~~ containment water level monitors or the. The applicant has not ~~completed~~ described his design for ~~this item~~ the. ~~Additional information is required.~~ This is an open item.

Position (6) requires containment hydrogen monitors. The applicant has not completed his design for this item. Additional information is required.

7.5.2.5 NUREG-0737 Item II.F.2 - Instrumentation for Detection of Inadequate Core Cooling

The applicant has not ~~completed~~ described his design for this item. ~~Additional information is required.~~ This is an open item.

7.5.2.6 Emergency Response Capability - R.G. 1.97 Rev. 2 requirements

~~NUREG-0737 Item II.F.3 - Instrumentation for Monitoring Accident Conditions (R.G. 1.97 Rev. 2)~~

Generic Letter No. 82-33 included additional clarification regarding Regulatory Guide 1.97, Revision 2 relating to the requirements for emergency response capability. In response to Generic Letter 82-33, the applicant referred to an earlier submittal of February 2, 1983 which describes the applicant's position on post-accident monitoring instrumentation. The staff has reviewed this response and ~~finds~~ found that there is insufficient information to complete the review of compliance to the regulatory guide. Additional information has been requested. ~~The applicant has not responded to staff's request, therefore,~~ this item remains open.

7.5.3 Evaluation Conclusion

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~~tion processing and~~
~~rameter display sys-~~
~~e staff could not~~
~~tion display system~~
~~ormation is required.~~

I think we should delete. DHFS has responsibility for SPDS — ICSB only has secondary review. (this is also included in the generic ltr on Em Response Capability)

7.6 Interlock Systems Important To Safety

7.6.1 Description

This section addresses the safety-related interlocks that

(1) prevent the overpressurization of low pressure systems

(2) prevent the overpressurization of the primary coolant system during low temperature operation

(3) ensure the availability of ECCS accumulators

(4) prevent an accidental startup of an isolated reactor coolant loop

The objectives of the review have to confirm that design considerations such as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of these safety-related systems.

7.6.2 Specific Findings

7.6.2.1 Residual Heat Removal System Isolation Valves Interlock

The residual heat removal (RHR) isolation valve interlocks are provided to prevent overpressurization of the RHR system. There are three motor operated valves in series in each of the two RHR pump suction lines from the reactor coolant system (RCS) hot legs. The two valves ~~are~~ located close to the containment penetration ~~walls~~, one outside ~~the containment~~ and one inside the containment, are provided with interlocks. The third valve inside the containment ~~has~~ is not interlocked and ~~The third valve~~ is operated by a keylock control switch which is under administrative control.

Two pressure transmitters powered from separate safety power trains are used for the isolation valve interlocks. Each valve is interlocked to prevent it from opening if RCS pressure is greater than 425 psig and to automatically close it if RCS pressure exceeds 700 psig. Valve position indications ~~is~~ are provided ~~for each valve~~ in the control room and at the auxiliary shutdown panel for each valve.

There are several inconsistencies in the FSAR ~~to describe~~ description of the RHR valve interlocks. Section 7.6.2.1 states that the pressure limit is 700 psig, but in Section 5.4.7.2.4 states that the limit is 750 psig. Figure 7.6-1 shows ~~some~~ additional interlocks ~~applied~~ on valve open circuits that is neither mentioned in Section 7.6.2.1 nor in Section 5.4.7.2.4. ~~The applicant has not addressed the~~ ~~The two pressure transmitters are not in diverse means that does not satisfy the Branch Technical Position ICSB-3 requirements. Additional information is required.~~ The applicant has not addressed the requirements of Branch Technical Position ICSB-3 for using diverse pressure transmitters. This is an open item.

7.6.2.2 Isolation of Low Pressure Systems From ~~The High~~ The High Pressure RCS ~~Pressure~~

General Design Criteria 15 requires that reactor coolant system and associated auxiliary, control and protection system shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation including anticipated operational occurrences. The staff requested that the applicant ~~to~~ identify all points of interface between the Reactor Coolant System (RCS) and ~~other~~ systems whose design pressure is less than that of the RCS and for each ~~such~~ interface to discuss the degree of conformance to the requirements of Branch Technical Position ICSB No. 3 ~~. Also discuss~~ and how the associated interlock circuits ~~circuitry~~ conforms to the requirements of IEEE 279. ~~The applicant has not responded to our request. Additional information is required.~~ This is an open item.

7.6.2.3 RCS Overpressure Protection ~~During Low Temperature Operation~~

The reactor coolant system overpressure protection during low temperature operation is ~~depended on~~ provided by the automatic opening of two pressurizer power operated relief valves (PORV). The actuation logic for PORV ~~is to~~ continuously monitors RCS temperature and pressure conditions. When pressure exceeds the programmed limit, an alarm will alert the operator to manually arm the system by ~~an ARM/Block~~ a switch located ~~in~~ on the main control board. During the design review, the staff raised a concern that a single failure could preclude the automatic actuation ~~logic~~ for all modes of operation including low temperature operation. The applicant ~~stated~~ noted that the design has not been finalized. ~~Additional information will be submitted by March 1984.~~ This is an ~~item remains~~ open item. ~~until additional information is reviewed and accepted by the staff.~~

7.6.2.4 Accumulator Isolation Valve Interlock

A motor-operated isolation valve is provided at each accumulator outlet. These valves are normally open during plant operation. To prevent an inadvertent closing of these valves, power is locked out from the valve motor control circuit. Administrative control is required to ensure that power is restored to the valve control circuit during plant shutdown. These valves are interlocked so that they

(1) open automatically on receipt of a safety injection signal.

(2) open automatically whenever the RCS pressure is above SI unblock (P-11) setpoint.

(3) cannot be closed as long as the safety injection signal is present.

Administrative controls require the performance of a periodic check valve leakage test. The interlock will ensure that the safety function is maintained during ~~the~~ testing.

During plant shutdown, the accumulator valves are in a closed position. To prevent an inadvertent opening of these valves during that period, the valve control power is locked out again. Administrative control is ~~required~~ used to ensure that these valves are closed during the prestartup procedures.

The accumulator motor operated valves are provided with indicating lights located at the control switches on ~~both~~ main control board and auxiliary shutdown panel. These lights are actuated by the valve motor operator limit switches. Another set of indicating lights is provided at the safeguard status panel. The status panel lights are actuated by a steam mounted valve position limit switch which is independent from the motor operator limit switches. The power source for indicating lights on the control panel

~~(previously you used just "status panel" & though the clarification need there be "safeguards")~~

is from 120 Vac Class 1E instrument power bus which is independent from the valve motor control power. The power source for status panel is from separate power supply. Therefore, the power lockout ~~at the control cir-~~ ~~cuit~~ will not affect either set of the indication lights. An alarm will sound when ~~the~~ a limit switch position senses that the valve is not fully open. The staff finds that the design satisfies the Branch Technical Position ICSB-4 and is, therefore, ~~is~~ acceptable.

The staff's evaluation on power lockout is based on information from schematic drawings provided during ~~the~~ a ~~ICSB~~ review meeting. The description in FSAR Sections 6.3.2.6, 6.3.5.5, and 7.6.3 ~~has some~~ is in errors since it ~~The FSAR~~ describes ~~that~~ the indicating lights ~~are~~ as powered by the valve control power ~~and the valve breakers are~~ which is removed during power lockout. ~~The staff requested the applicant to correct these errors in the FSAR.~~ This item is confirmatory subject to ~~corrected documentation~~ proper revision in the FSAR to update the description to eliminate these errors.

7.6.2.5 Reactor Coolant System Loop Isolation Valve Interlocks

The FSAR Section 7.6.5 describes the reactor coolant system loop isolation valve interlocks. The description is ~~not clear and~~ is incomplete and additional information is required to clarify that the design is in conformance with IEEE-279 requirements. ~~Administrative control~~ may be required because ~~partial loop operation~~ has not been licensed by the NRC.

This is an open item.

7.6.3 Evaluation Conclusion

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7.7 Control Systems

7.7.1 Description

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The plant control systems ~~that~~ are not relied on to perform safety functions but ~~that~~ control plant processes while hav~~ing~~e an impact on plant safety are described in this section and include the following:

(1) reactor control system

(2) rod control system

(3) monitoring and indicating system

(4) plant control interlocks

(5) pressurizer pressure control

(6) pressurizer level control

(7) steam generator water level control

(8) steam dump control

(9) Incore instrumentation

(10) Boron concentration measurement system

The reactor control system enables the nuclear plant to follow load changes automatically including a 10% step change in load ~~increase or decreases~~ ~~and~~ or a 5% per minute ~~rate of load~~ ~~ramp increase or decrease.~~ change. The system ~~also can~~ maintain ~~restore~~ coolant average temperature following a load change. The reactor control system controls the reactor coolant average temperature within ~~to~~ acceptable limits ~~value~~

by regulation of control rod bank position. The core axial power distribution is controlled manually during load following maneuvers by changing the boron concentration in the reactor coolant system.

The rod control system provides for reactor power modulation by manual or automatic control of control rod banks in a preselected sequence. It displays control rod positions, alerts the operator in the event of control rod deviation exceeding a preset limit, and alerts the operator on inadequate shutdown margins resulting from excessive control rod insertion. The automatic rod control system is designed to maintain a programmed average temperature in the reactor coolant by regulating the reactivity within the core. The automatic rod control is performed between 15 and 100% of rated power. Power to rod drive mechanisms is supplied by two motor generator sets operating from two separate 480V three phase buses. Each generator is the synchronons type and is driven by a 200 Hp induction motor. The AC power is distributed to the rod control power cabinets through the two series connected reactor trip breakers. The reactor trip breakers are part of the safety system as described in Section 7.2 of this report.

The monitoring and indicating systems include:

(1) Nuclear instrumentation monitoring

  (a) Nuclear power level

  (b) axial flux imbalance

  (c) Upper radial tilt

  (d) Lower radial tilt

(2) Rod position monitoring

  (a) digital rod position indication

  (b) demand position

  (c) rod insertion limits

  (d) rod deviation alarm and rod bottom alarm

The plant control interlocks prevent further withdrawal of the control rod banks either by a control system malfunction or an operator error. The interlocks are derived from nuclear instrument channels or reactor coolant overtemperature-overpower channels. The interlocks also limit automatic turbine load increases during a rapid return to power transient (through the negative moderator coefficient). The interlock can be cleared by an increase in coolant temperature, which is accomplished by reducing the boron concentration in the coolant.

The reactor coolant pressure is controlled by using either the heaters or the spray ~~of the~~ pressurizer plus PORV steam relief for large transients. The water inventory in the RCS is maintained by the CVCS. During normal plant operation, the charging flow varies to match the flow demanded of the pressurizer water level controller. The pressurizer water level is programmed as a function of coolant average temperature. During startup and shutdown operations, the charging flow is manually regulated to maintain pressurizer water level.

The steam generator level is programmed by a three-element feedwater controller, which regulates the feedwater valves by continuously comparing the feedwater flow signal, the water level signal, the programmed level setpoint, and the steam flow signal. During startup or low power operation, a feed-forward control scheme uses steam generator level and nuclear power signals to position a bypass control valve, which is in parallel with the main feedwater regulating valve.

The steam dump system is designed to accept a 50% load rejection without tripping the reactor. The system functions automatically by bypassing steam directly to the condenser and/or atmosphere to maintain ~~an artificial~~ The load on the primary system. The rod control system can then reduce the reactor coolant temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions.

A demand signal for the load-rejection steam dump controller is generated if the difference between the reference reactor coolant average temperature (based on turbine impulse chamber pressure) and the measured reactor coolant average temperature exceeds a preset value.

The incore instrumentation system consists of chromel-alumel thermocouples at fixed core outlet positions and movable miniature neutron detectors at selected fuel assemblies. The thermocouple readings are monitored by the plant computer. The movable detectors can perform flux mapping at various core quadrant to obtain a flux

location

map for any region of the core. The data collection, calculation and recording are performed by the plant computer.

The boron meter determines the relative concentration of ~~the~~ boron ~~isotope~~ in the sample fluid. The boron concentration measurement system is designed for use as an operating aid. ~~It is not designed as a safeguard system~~.

7.7.2 Specific Findings

7.7.2.1 Control System Failures Caused by Malfunctions of Common Power Source or Instrument Line

To provide assurance that the FSAR Chapter 15 analyses adequately bounds events initiated by a single credible failure or malfunction, the staff ~~has~~ asked the applicant to identify any power source or sensors that provide power or signals to two or more control functions, and demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences more severe than those of Chapter 15 analyses or beyond the capability of operator or safety systems.

The applicant has not provided a respon to this open item. ~~Additional information is required~~.

7.7.2.2 Control System Failures Caused By High-Energy Line Breaks

Operating reactor licensees were informed by IE Information Notice 79-22~~, issued September 19, 1979,~~ that if certain nonsafety-grade control equipment were subjected to the adverse environment of a high energy line break, it ~~could~~ may impact the safety analyses and the adequacy of the protection functions performed by the safety-grade equipment. The staff has requested a review ~~by the applicant~~ to determine whether the harsh environment associated with high-energy line breaks might cause control system malfunction and result in a consequence more severe than those of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

The applicant has not provided a response to this open item. ~~Additional information is required~~.

7.7.2.3 Freeze Protection For Instrument Sensing Lines

~~The staff concerns that the~~ Instrument sensing lines that can be exposed to freezing temperature ~~should be~~ are provided an environmental control system (heating and ventilation or heat tracing) to protect the lines from freezing during extremely cold weather. The environment associated with safety related sensing lines should be monitored and alarmed so that appropriate corrective action can be taken to prevent loss of or damage to the lines from freezing in the event of loss of the environmental control system. The staff requested the applicant to document the freeze protection system design in Section 7.7 of the FSAR. ~~Additional information is required.~~ This is an open item

7.7.2.4 <u>NUREG-0737 Item II.K.3.9, Proportional Integral Derivative (PID) Controller Modification</u>

Westinghouse recommended that the derivative time constant in the pressurizer PORV PID controller be set to off to address this action Plan item. This action removes the derivative action from the controller so that the actuation signal to this valve is no longer sensitive to the rate of change of pressurizer pressure. The applicant has implemented this recommendation. The staff finds that the applicant~~'s~~ is in compliance with the Action Plan guidelines for this item, ~~is acceptable.~~

7.7.3 <u>Evaluation Conclusion</u>

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