UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 0 0 1983

MEMORANDUM FOR: Darrell G. Eisenhut, Director, Division of Licensing

FROM: Roger J. Mattson, Director, Division of Systems Integration

SUBJECT: RESPONSE TO DIABLO CANYON ALLEGATIONS

Enclosed is the Division of Systems Integration (DSI) response to those Diablo Canyon allegations identified in George Knighton's memorandum dated November 29, 1983 as DSI responsibility. We have provided responses to Allegation Nos. 3, 4, 5, 6, 37, 39, 40 and 45 in accordance with the evaluation guidance provided in George Knighton's memorandum dated November 30, 1983. As discussed and agreed previously with George Knighton, Allegation Nos. 11, 12 and 13 were incorrectly assigned to DSI (Auxiliary Systems Branch) and therefore, no input is included for these concerns.

Roger J. Mattson

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Task: Allegation or Concern No. 3

ATS No.: NRR-83-02 BN No.: 83-03(1/7/83)

Characterization

A concern was raised that the pressure boundary of the nonessential loop of the safety-related component cooling water system (CCWS) although not required to function following a safe shutdown earthquake (SSE) was not qualified for the SSE. This loop would therefore fail in an SSE resulting in loss of water and subsequent CCWS failure when a single active failure (to close) is assumed in the isolation valve to the nonessential loop.

Implied Significance to Plant Design, Construction or Operation

The potential loss of the CCWS as postulated in the concern would affect the ability to safely shutdown the plant following an SSE. Therefore, a reanalysis of the CCWS seismic qualification design and associated modifications to ensure its functional integrity following an SSE would be required.

Assessment of Safety Significance

The staff has evaluated the concern against the CCWS design information available in the FSAR and against information obtained in subsequent correspondence and meetings held with the licensee. The staff has verified that the CCWS including the pressure boundary of the nonessential loop is qualified to the SSE, and, therefore, no system failure of the type postulated in the concern should occur. That is, a postulated single active failure (to close) in the nonessential loop isolation valve will not result in an unacceptable condition in the CCWS because isolation of the nonessential loop following an SSE is not essential for ensuring the CCWS safety function. Refer to Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement No. 16 for further information.

Staff Position

The staff concludes that the CCWS design satisfies General Design Criteria 2 and 44 with respect to assuring its cooling water safety function following an SSE and concurrent single active failure. This concern has been satisfactorily resolved.

Action Required

None

Task: Allegation or Concern No. 4

ATS No.: NRR 83-02 BN No.: 83-03(1/7/83)

Characterization

A concern was raised that a single failure (to close) in the isolation valve to the nonessential loop of the component cooling water system (CCWS) concurrent with a loss of coolant accident (LOCA) would result in an increase in the heat load on the CCW heat exchangers beyond their design heat removal capability because of failure to isolate nonessential heat loads.

Implied Significance to Plant Design, Construction or Operation

The potential inability of the CCWS to remove sufficient heat following a LOCA with a subsequently resulting higher than design allowable CCWS temperature could cause a failure of safety-related equipment to perform their function. Therefore, an evaluation of the consequences of this postulated occurrence with verification of satisfactory CCWS heat removal performance was required.

Assessment of Safety Significance

The staff has evaluated the concern against the CCWS design information available in the FSAR and against information obtained in subsequent correspondence and meetings held with the licensee. The staff has verified that the postulated event (LOCA with a concurrent single failure to close in the nonessential loop isolation valve) does not result in a heat load in excess of the design heat removal capability of the CCWS heat exchangers. Refer to Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement No. 16 for further information.

Staff Position

The staff concludes that the CCWS design satisfies General Design Criterion 44 with regard to assuring its cooling water safety function under the above assumed condition. This concern has been satisfactorily resolved.

Action Required

None

Task: Allegation or Concern No. 5

ATS No.: NRR 83-02 BN No.: 83-03(1/7/83)

Characterization

A concern was raised that with all redundant essential heat loads imposed on the component cooling water system (CCWS) following a loss of coolant accident (LOCA), the CCWS could not remove sufficient heat to maintain the design maximum CCWS temperature and assure a safe shutdown. This is because only one CCW heat exchanger is normally on line and operator action could not be taken soon enough to align the normally isolated redundant CCW heat exchanger prior to exceeding the allowable CCW temperature.

Implied Significance to Plant Design, Construction or Operation

The potential inability of the CCWS to remove sufficient heat following a LOCA with a subsequently resulting higher than design allowable CCWS temperature could cause a failure of safety-related equipment to perform their function. Therefore, an evaluation of the consequences of this postulated occurrence with verification of satisfactory CCWS heat removal performance was required.

Assessment of Safety Significance

The staff has evaluated the concern against the CCWS design information available in the FSAR and against information obtained in subsequent correspondence and meetings held with the licensee. The results of this review indicated that the originally assumed ultimate heat sink (Pacific Ocean) temperature of 70°F was too high for adequate heat removal following a LOCA with all essential equipment operable and one CCW heat exchanger on line assuming a concurrent single failure in an auxiliary salt water pump. Under this limiting

condition from the standpoint of CCWS heat removal capability, a maximum ocean water temperature of 64°F must be assumed in order to assure that the design allowable CCWS temperature is not exceeded. The licensee has proposed a technical specification which requires that the redundant CCW heat exchanger be aligned whenever the ocean water temperature exceeds 64°F. Otherwise, the plant must be shutdown. The staff has accepted this technical specification and it has been incorporated in the Plant Technical Specifications. Refer to Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement No. 16 for further information.

Staff Position

The staff concludes that with incorporation of the above technical specification limit on CCWS operation that the CCWS design satisfies General Design Criterion 44 with regard to assuring its cooling water safety function under design basis accident conditions. This concern has been satisfactorily resolved.

Action Required

None

DIABLO CANYON ALLEGATIONS

Task: Allegation #6a

ATS No.: NRR 83-02 BN No.: 83-03 (1/7/83)

Characterization

Instrumentation and controls required to perform safety related functions do not conform to Seismic Category 1 requirements (e.g., component cooling water system surge tank level instrumentation).

Implied Significance to Plant Design, Construction, or Operation

In accordance with General Design Criterion 2 (Design Bases for Protection Against Natural Phenomena); the Diablo Canyon accident analysis assumes the proper functioning of instrumentation and controls used to mitigate the effects of accidents in conjunction with the effects of natural phenomena such as earthquakes. Instrumentation and controls, that are 1) relied upon to perform safety functions and 2) not seismically qualified, cannot be assumed to function following a seismic event.

Assessment of Safety Significance

The component cooling water system (CCWS) at Diablo Canyon consists of three loops, A, B, and C. The CCWS is a Design Class 1 system except for non-vital components in loop C. An analysis has been performed by PG&E to demonstrate

that the non-vital components will not fail as the result of a design basis seismic event (i.e., a safe shutdown earthquake; SSE). The CCWS surge tank is seismically qualified and is divided into two separate volumes to provide redundancy, thereby ensuring adequate cooling water to safety related loads following an accident. The surge tank level instrumentation is seismically qualified from a pressure boundary standpoint, however, it has not been shown to function properly following a seismic event. The surge tank level instrumentation is used to automatically provide water from the Makeup Water System to the surge tank in the event of a low level. Since the CCWS is seismically qualified and therefore surge tank level is not expected to change during a seismic event, and since the surge tank level instrumentation is not used to perform a safety function, it need not be seismically qualified from an operational standpoint. Therefore, the staff concludes that this instrumentation is acceptable.

The licensee has stated and the Diablo Canyon SER notes that instrumentation and control components required to perform a safety function are designed to meet seismic Category 1 requirements. In accordance with the Standard Review Plan (SRP), the staff reviews the instrumentation and controls taken credit for in the accident analysis to assure they have been appropriately classified (i.e., as required to perform a safety function). Subsequent independent design reviews also verify the proper design classification of instrumentation and control components. Based on these reviews, the staff finds the instrumentation and controls at Diablo Canyon to be acceptable.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

DIABLO CANYON ALLEGATIONS

Task: Allegation #6b

ATS No.: NRR-83-02 BN No.: 83-03 (1/7/83)

Characterization

Instrumentation and controls used to isolate main feedwater flow following a main steamline break are not safety related (i.e., do not conform to Class 1E and seismic requirements).

Implied Significant to Plant Design, Construction, or Operation

The failure to isolate main feedwater flow following a main steamline break could result in an energy (steam) release to containment greater than assumed in the accident analysis (FSAR Chapter 15); the concern here is over-pressurization of the containment structure. Failure to isolate could also result in an additional (unwanted) cooldown of the reactor coolant system causing a reduction of core shutdown margin not considered in the accident analysis.

Assessment of Safety Significance

Isolation of main feedwater following a steamline break is initiated by the Engineered Safety Features Actuation System (ESFAS). Isolation is accomplished by closing all main control valves, tripping the feedwater pumps, and closing the feedwater isolation valves. The feedwater isolation valves (also referred to as the backup feedwater isolation valves) are Category I containmeny isolation valves, i.e., they are designed to Class 1E requirements, including seismic qualifications. The ESFAS is also designed to Class 1E requirements, including seismic qualifications.

Physical separation is maintained between redundant ESFAS circuits, including field wiring. The tripping of the main feedwater pumps and closure of the feedwater control valves are redundant to closure of the safety Class 1 feedwater isolation valves and are not necessary for safety. The feedwater isolation valves, as well as all other containment isolation valves, were included in the PG&E Systems Interaction Program. The Diablo Canyon accident analysis for a main steamline break concludes that there is no consequential damage to the primary system or the core, and that there is no failure of the containment structure. The staff agrees with this assessment.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

DIABLO CANYON ALLEGATIONS

Task: Allegation No. 37

ATS No.: RV 83A41 BN No.: 83-169 (10/20/83)

Characterization

The solid state protection system (SSPS) relays that initiate closure of RHR letdown isolation valves 8701 and 8702 perform no safety function, reduce the reliability of the RHR system, and cause a potential for RHR pump damage. Therefore, these relays should be removed.

Implied Significance to Plant Design, Construction, or Operation

The RHR letdown line contains two isolation valves (8701 and 8702) in series that are normally closed during power operation. These valves are opened when entering Mode 4 (hot shutdown) to allow the RHR pumps to take suction from the reactor coolant system (RCS) to the RHR heat exchangers for decay heat removal. Both valves 8701 and 8702 are interlocked so that they will automatically close to isolate the RHR system from the RCS if RCS pressure increases to a pre-determined setpoint. This automatic isolation function (performed by the Westinghouse designed SSPS) is provided to protect the low pressure RHR system piping from higher RCS pressures. Isolation is accomplished using a "fail safe" design (i.e., on a loss of SSPS power, valves 8701 and/or 8702 will automatically close). The concern here is that a loss of SSPS power will cause an unwanted (spurious) isolation of the RHR letdown line causing eventual RHR pump damage assuming no operator action.

Assessment of Safety Significance

Isolation of the low pressure RHR system from the high pressure RCS must be provided to prevent RHR system overpressurization that could potentially result in a loss of coolant accident (LOCA) outside containment. Therefore, RHR letdown line isolation is a safety function. The SSPS, including relays, which performs this function is safety related and designed to Class 1E requirements. Both valves 8701 and 8702 are provided with this automatic closure interlock on increasing RCS pressure so that a single failure will not prevent RHR letdown line isolation. Therefore, the relays used to initiate closure of these valves are essential and should not be removed.

Diverse indications and alarms are provided in the control room (including a RHR system low flow alarm to be installed during the first refueling outage) to allow the operator(s) to assess RHR system status and to alert them to potential system degradation. Technical Specification surveillance requirements at Diablo Canyon include periodic verification of RHR system flowrate when using the RHR letdown line. In addition, diverse means of decay heat removal (i.e., reactor coolant loops) can be readily made available should the RHR letdown line be inadvertently/spuriously isolated.

Based on the above, the staff concludes that the existing SSPS design regarding RHR letdown line isolation is acceptable.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

DIABLO CANYON ALLEGATIONS

Task: Allegation #39

ATS No.: RV 83A47 BN No: 83-169 (10/20/83)

Characterization

There is no control room annunciation provided to alert the operator(s) when the RHR letdown line has been isolated during Modes 4, 5, and 6 (hot shutdown, cold shutdown, and refueling respectively).

Implied Significance to Plant Design, Construction, or Operation

During modes 4, 5, and 6 the residual heat removal (RHR) system is aligned in the shutdown cooling mode by taking suction from reactor coolant system (RCS) loop 4 through the RHR letdown line to the RHR pumps. The RHR pumps direct flow through the RHR heat exchangers for decay heat removal via the component cooling water (CCW) system, and then back to the RCS cold legs. There are two isolation valves (8701 and 8702) in series located in the RHR letdown line. If one of these valves should inadvertently close, RHR pump suction would be lost. The concerns here are loss of decay heat removal capability and potential damage to the RHR pumps. It has been estimated that pump damage could occur as soon as 10 to 15 minutes following a spurious isolation of the RHR letdown line, if the RHR pump is not shut off.

Assessment of Safety Significance

For those modes of operation where RHR shutdown cooling is used, only one RHR train or one filled reactor coolant loop is necessary to provide sufficient decay heat removal capability. The Diablo Canyon Technical Specifications require either two RHR trains to be operable and/or two filled reactor coolant loops to be available in order to allow for single failures. If both RHR trains are being used and the RHR letdown line becomes isolated, the operator(s) would have sufficient time to fill at least one coolant loop (assuming no loops are filled) for decay heat removal. Control room indications of loss of decay heat removal include RCS temperature, RHR system flow, and RHR pump discharge pressure. With less than the required number of reactor coolant loops and/or RHR trains operable, the Technical Specifications require immediate corrective actions to return the required loop/train to operable status as soon as possible.

Indication provided in the control room of RHR letdown line isolation includes position indication for valves 8701 and 8702 (red and green position status lights next to the valve control switches on the main control board) as well as RHR system flow, pressure, and pump status information. Although these features do provide a capability to assess RHR system status, the staff has recognized the need for installation of a RHR low flow alarm. Accordingly,

the licensee is required to install a RHR low flow alarm during the first refueling. This requirement is documented in Supplement No. 13 of NUREG-0675, "Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2." The staff has concluded that the existing control room indications and procedures are sufficient to assure adequate decay heat removal in the interim.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

DIABLO CANYON ALLEGATIONS

Task: Allegation #40

ATS No.: RV83A 47 BN No.: 83-169 (10/20/83)

Characterization

The question raised was with regard to whether or not the single RHR pump suction line from the RCS hot leg meets safety related standards. The newer PWRs are designed with redundant RHR pump suction lines from the RCS hot legs.

Implied Significance to Plant Design, Construction or Operation

The RHR suction line from the RCS hot leg in Diablo Canyon contains two isolation valves (8701 and 8702) in series that are normally closed during power operation. When the RHR system is operated as a part of the ECCS, the RHR pump suctions are aligned with either the RWST or the containment emergency sumps. The RHR suction line from the RCS hot leg is only used during modes 4 (hot shutdown while RCS temperature is less than 323°F), 5 (cold shutdown) and 6 (refueling). A postulated failure of either isolation valve (8701 or 8702) in the RHR suction line to open during plant shutdown could prevent the plant from reaching a cold shutdown condition.

Assessment of Safety Significance

In the Diablo Canyon SER Supplement No. 7, the staff stated that the single RHR suction line from the RCS hot leg was acceptable. The staff conclusion was based on the following:

(1) The Diablo Canyon design has a safety related Auxiliary Feedwater System (AFWS). The condensate storage tank is the primary source of AFW with about an 8 hour water supply. In order to ensure the capability to remove heat via the steam generators for extended periods, provisions have been made to connect the raw water reservior to the suction line of the AFW pump. This will provide enough AFW to allow an additional 100 hours of steam generator operation for both units.

(2) The licensee has indicated that the combination of a mechanical failure of the RHR isolation valves and an earthquake results in a risk of about 10% of the core melt risk from all causes calculated in the Reactor Safety Study.

Branch Technical Position RSB 5-1 was not approved at the time SSER No. 7 for Diablo Canyon was issued. In accordance with the implementation schedule of BTP RSB 5-1, the Diablo Canyon Units are considered class 2 plants which are not required to fully implement this BTP. Table 1 of BTP RSB 5-1 shows what is necessary to be implemented for class 2 plants. A single RHR suction line from the RCS hot leg is considered acceptable for a class 2 plant as long as a single failure could be corrected by manual actions inside or outside of containment, or the plant could be returned to hot standby until manual actions (or repairs) are accomplished. (page 5.4.7-16 of SRP 5.4.7). Also, BTP RSB 5-1 for class 2 plants requires that the RHR isolation valves have independent,

diverse interlocks to protect against one or both valves being open during an RCS pressure increase above the design pressure of the RHR system. There was no assessment of the degree of compliance of the Diablo Canyon design against BTP RSB 5-1 documented in any staff SSER.

Based on the above facts, the staff evaluation of the subject allegation is as follows:

The RHR suction line from the RCS hot leg is not required for ECCS functionability. The RHR pumps take suction from RWST or containment emergency sumps, and serve the ECCS function during a LOCA. The suction line from RCS hot leg is used only for modes 4 (< 323°F), 5 and 6. GDC 34 of Appendix A to 10CFR 50 requires that the decay heat removal safety function should be accomplished assuming a single failure. The Diablo Canyon design complies with this requirement by having a RHR system plus a safety related AFWS (with steam generators and atmospheric steam dump valves). Based on the above, we conclude that the Diablo Canyon design meets GDC 34 and the intent of BTP RSB 5-1. The current RHR design is acceptable for safe operation at Diablo Canyon.

The staff is currently conducting a generic evaluation of the requirements for shutdown decay heat removal systems. This work is being performed as an Unresolved Safety Issue (TAP A-45), and the staff's assessment is projected to be complete within one year. Final resolution of this issue may or may not result in additional requirements.

Staff Position

This allegation does not involve considerations not previously considered for plant readiness for power ascension testing or full power operation.

Action Required

None

DIABLO CANYON ALLEGATIONS

TASK: Allegation #45

ATS NO.: RV 83A47 BN NO.: 83-169 (10/20/83)

Characterization:

Section 5.5. of the Diablo Canyon FSAR describes the autoclosure interlock for the RHR Suction line isolation valves (8701 and 8702). Section 3.4.9.3.a of the Diablo Canyon Technical Specifications requires power to be removed from these isolation valve operators during modes 4 (Hot shutdown when RCS cold leg temperature is less than 323°F), 5 (cold shutdown) and 6 (refueling). This requirement defeats the function of autoclosure interlock for the valves.

Implied Significance to Plant Design, Construction or Operation

As the result of Technical Specification Section 3.4.9.3.a, the isolation valves (8701 and 8702) will be left in an open position with power removed during low pressure/temperature operation of the plant. The automatic closure interlock to these isolation valves causes them to lose their design function. This will result in a situation in which there is in sufficient isolation capability feature to prevent an intersystem LOCA between the high pressure RCS and the low pressure RHR system.

Assessment of Safety Significance

Section 5.5 of the Diablo Canyon FSAR states that during low pressure/temperature operation, the isolation valves (8701 and 8702) between the RCS and the suction of the RHR pumps are interlocked with a pressure signal to automatically close the valves whenever the RCS pressure increase above approximately 600 psig. Section 3.4.9.3.a of the Diablo Canyon Technical Specification requires the RHR system isolation valves

(8701 and 8702) to be open with power removed from the valve operators while the positive displacement charging pump is in operation. The applicability of the T.S. is during mode 4 when the temperature of any RCS cold leg is less than or equal to 323°F, mode 5, or mode 6 with the reactor vessel head on this Technical Specification requirement defeats the automatic closure interlock function as designed.

Power removal from valves 8701 and 8702 while the RHR system is operating was required by the staff as the result of a meeting with the licensees on RCS low temperature overpressure protection (LTOP) and RHR pump protection concerns. Since the Diablo Canyon design has only one RHR suction line from the RCS, a spurious automatic closure of the isolation valve would result in loss of RHR pump suction flow and would result in a RCS pressurization as a result of the loss of letdown flow. However, there was no documentation (SSER, letter or meeting minutes) of the staff's basis for requiring power removal from those isolation valves during modes 4, 5 and 6.

In the Diablo Canyon SER Supplement No. 13, section 6.3. (ECCS), dated April 2, 1981, the staff concluded that the licensee should be required to provide an alarm to alert the operator to a degradation in ECCS (during long term recirculation). A low flow alarm was stated to be an acceptable method to satisfy this concern and the staff indicated that an alarm should be installed at the first refueling outage. Until then, procedures and dedicated operators were to be implemented during long term recirculation to manage and monitor ECCS performance. There was no documentation to indicate that the licensee committed to this

staff position, nor was this staff position included in the Diablo Canyon low power license. SRP 5.4.7 (BTP RSB 5-1) requires an autoclosure interlock on the RHR suction line isolation valves. Without power to the valve operators, the autoclosure function is defeated.

Based on the above facts, the staff evaluation of the subject allegation is as follows:

Without power to the isolation valve operators, the plant design does not conform to BTP RSB 5-1, Position B.1.C, for the requirement of autoclosure interlock. By having power available to the isolation valves during shutdowns ensures an event V (intersystem LOCA) will not occur as a result of the operator failing to close both isolation valves during a return to power.

With power on the isolation valves, a spurious closure of the isolation valves would result in a loss of suction flow to the RHR pumps. However, the low flow alarm discussed in SSER No. 13 would enable rapid operator detection and mitigation. The licensee has informally indicated that a minimum of 10 minutes without adequate suction pressure would be available without pump damage. Also, there are numerous indications available to alert the operator to improper RHR valve alignment (A list is provided in staff evaluations to allegation No. 37 and 39).

Staff Position

To implement the staff position stated in SSER No. 13, the installation of a low flow alarm for RHR pump protection is being considered as a

license condition in the Diablo Canyon full power license. Additionally, it is the staff position that power be available to the RHR MOVs when in a shutdown condition. However, there is a question as to when these requirements should be implemented. If the low flow alarm were not installed until the first refueling outage, reinstating power to the RHR MOVs in the meantime would result in the autoclosure interlock being enabled to provide protection against intersystem LOCA.

However, the chances of spurious autoclosure and consequent loss of RHR suction pressure (without the low flow alarm) and of an overpressure event would be increased. If power restoration to the RHR MOVs were not implemented until the low flow alarm is installed at the first refueling outage, the chance of loss of RHR suction in the interim is reduced but there is a possibility of an intersystem LOCA. To determine which option results in the safest operation of the plant, the staff considered the following:

1. During the first cycle of operation, plants operate more frequently on the RHR system as a result of maintenance, testing and training requirements for a new plant. Thus, the period of vulnerability to a spurious RHR suction MOV closure may be greater than in subsequent cycles.
2. The RHR relief valve would open to relieve pressure if a plant startup were attempted with both RHR MOVs open. It is not, in the staff's judgment, credible to postulate plant startups with both MOVs left open. The operator would have to shut at least one MOV to continue the plant startup.
3. Failing to close the second RHR suction MOV would not, in itself, result in an intersystem LOCA. The first MOV must also fail. The

first MOV can fail in either of two ways by either the "open permissive" interlock failing along with the operator reinstating power to the valve, (it is required to be de-energized§ed) them attempting to open the valve. The second mode of failure would be for the valve to rupture in such a way that flow between the two systems occurred. Both of these two failure modes are judged to have an extremely low probability. However, the consequences of an intersystem LOCA could be severe.

4. There have been many occasions if spurious RHR suction valve closures on operating plants. This has resulted in not only a loss of decay heat removal, but also an overpressure event due to the loss of the letdown flowpath.

Based on the above factors, the staff believes the best course of action is to continue the current technical specification for power to be removed from the RHR MOVs during Modes 4, 5 and 6 until the low flow alarm is installed. However, the staff position that would permit the licensee to wait until the first refueling outage before installing the low flow alarm was taken over two years ago. We will puruse with the licensee a commitment to a schedule for accomplishing this installation at the earliest possible time. In the interim, until the low flow alarm is installed, the staff believes that strict administrative controls should be developed and implemented to ensure that MOVS 8701 and 8702 are closed with power removed during plant startups when RCS pressure is above the RHR design pressure.