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Licensee: Connecticut Yankee Atomic Power Company
P. O. Box 270
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Facility Name: Haddam Neck Nuclear Power Plant
Audit At: Haddam Neck, Connecticut
Audit Conducted: July 24 - August 4, 1989

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Audit Summary: Audit on July 24 - August 4, 1989

Areas Audited: Special announced audit of equipment, activities, and human performance which are important to prevent, mitigate, or recover from the ISLOCA events and subsequent core melt. Specifically, five systems with postulated ISLOCA events: RHR System; High Pressure Injection System; Core Deluge/Low Pressure Injection System; Chemical & Volume Control System; Alternate Letdown/Drain Line were included. The audit included 720 inspector hours on-site and 150 hours off-site and at the NRR Headquarters Office.

Results: No regulatory findings were identified, and several concerns relevant to potential ISLOCA events were identified. These observations were beyond normal compliance boundary.

ISLOCA AUDIT REPORT

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EXECUTIVE SUMMARY

This report documents the results of an announced team audit performed at Connecticut Yankee Atomic Power Company, Haddam Neck plant from July 24, 1989 through August 4, 1989. The audit team examined plant hardware and operational activities relating to high-and-low pressure Interfacing Systems LOCA (ISLOCA). Emphasis was placed on man-machine interactions as well as human performance.

BACKGROUND

The Interfacing System LOCA refers to a class of loss-of-coolant accidents in which the high pressure boundary of the Reactor Coolant System, interfacing with low pressure piping, is breached. A special concern is the overpressurization of low pressure systems which may result in a rupture outside of the containment and thus discharge reactor coolant to the environment. Furthermore, LOCA event mitigation systems, such as Emergency Core Cooling Systems, and other injection paths may be affected, resulting in a core melt.

ISLOCA accidents of these types, referred to as Event "V" in the Reactor Safety Study of WASH-1400, may lead to significant radiological releases. However, the probability and consequence of the event are heavily dependent on plant features, break locations and mitigating actions, and are subject to substantial uncertainties.

Because of the risk potential and the uncertainties, NRR has established an ISLOCA program with the following goal;

> High confidence that a high consequence ISLOCA will not occur for the present generation of reactors.

To assure that the goal is met, the following areas will be explained:

(1) The likelihood that an ISLOCA will occur.

(2) The likelihood that core damage as a result of an ISLOCA can be prevented or significantly delayed with relaince on existing plant equipment, procedures, and training.

(3) The likelihood that, in case of an ISLOCA leading to core melt, there exist provisions of equipment, procedures, and training to minimize the offsite radiological consequences.

The ISLOCA program effort will include a series of audits at selected plants. The purpose of these audits is to obtain information that would enhance the understanding of the various aspects and factors affecting the event frequency and severity of an event. Information generated through the audit process will be used as an input to develop PRA modeling of ISLOCA events, and may provide a valuable input for future development of regulatory guidance. Furthermore, the findings of this audit may serve as valuable input to validate modeling methodology and techniques of Human Reliability Analysis (HRA) as well as quantification of human actions as related to ISLOCA events.

OBJECTIVES AND SCOPE

The primary objective of this audit was to gather facts and collect data on the "AS BUILT" and "AS FOUND" plant conditions, including design features, systems, equipment, procedures, operations and human performance as related to ISLOCA. Specific tasks of the audit team included identification of generic events or features as related to ISLOCA, principal systems and their interactions, potential initiating or precursor events, and human performance and potential human errors.

Within the scope of the primary objective, the audit also included an assessment of licensee programs relevant to ISLOCA, and reviewed various licensee records to determine what preventive, corrective and mitigation measures were in place and if they were adequate. Also, that the availability of equipment and systems important to ISLOCA is adequate. The audit team further observed and identified critical elements of human factors and potential errors (human performance) to prevent, detect, or recover, should an event or symptoms of an event occur.

Five systems were selected on the basis of postulated ISLOCA initiations, event scenario, and potential consequences. A total of twelve (12) ISLOCA events were postulated from the five systems for audit activities. The five systems were:

- o Residual Heat Removal (RHR)
- o High Pressure Injection System (HPI)
- o Low Pressure Injection(LPI)/Core Deluge (CD) system
- o Letdown Line of Chemical and Volume Control System (CVCS)
- o Alternate Letdown Drain System

The licensee classified 22 Valves as category 'A' Pressure Isolation Valves (PIVs) which interface between high and low pressure systems. An additional 8 valves, identified by the audit team, were included in the audit on the basis of their potential contribution to prevention, mitigation, or recovery from an ISLOCA event.

The "AS BUILT" design features were reviewed to identify the potential inadvertent overpressurization of low pressure interfacing systems and components by reviewing various documents and records pertinent to ISLOCA. The "AS IS" and "AS FOUND" conditions, including human performance and human factors such as man-machine interface, were evaluated. Also, emphasis was placed on the awareness of plant personnel concerning the potentials of ISLOCA and licensee initiatives to prevent, detect, mitigate or recover from an ISLOCA event.

AVAILABILITY OF EQUIPMENT

The availability and integrity of plant equipment was evaluated to assure that they would operate in accordance with their intended safety functions should their services be demanded. Accordingly, the plant maintenance program was evaluated to assure that the preventive measures, corrective maintenance, routine work controls (including jumpers, tagging and work orders), and periodic surveillances were addressed and performed effectively. Thus, the conduct of the audit included evaluation of station maintenance activities to ascertain that they are performed adequately and effectively in accordance with prescribed written procedures, and that generic problems and recurring failures of equipment were adequately addressed in the station maintenance, Inservice Test (IST), and surveillance programs. To assess the implementation of the programs, "AS FOUND" states of the equipment were evaluated by performing "walkthroughs," i.e., visual inspections, witnessing of in-progress activities, use of mockups, and "hands on" simulation. The effectiveness of the preventive and corrective maintenance measures were evaluated by reviewing appropriate work records and the performance trending of equipment.

Subjective observations were made to assess the accessibility of plant equipment for manual emergency operations. The environmental and radiological conditions were evaluated in conjunction with the accessibility for manual operation of system PIVs and MOVs should an ISLOCA occur and such local operations be demanded. Surveillance and IST records were reviewed to identify the plant vulnerability in terms of likelihood of ISLOCA events or potential precursor events.

HUMAN PERFORMANCE

To assure plant risk is minimized, a high degree of equipment availability must be complemented by the ability of the plant staff to operate without the introduction of human errors, and to detect, respond or recover from an event. Reliability of human performance, thus, includes familiarity of plant staff with plant procedures and equipment, and capability of performing routine, normal, abnormal and emergency operations without introducing human errors. The conduct of the audit focused on the extent to which the design and operation of the plant influences human performance, relative to the identification of potential precursor, detection, diagnosis and mitigation of ISLOCA events.

A credible series of plant evolutions were identified on the basis of the postulated ISLOCA scenarios, and specific operator decision-action sequences that would contribute to the course and outcome of an event were evaluated. Particular emphasis was placed on the awareness of ISLOCA events by plant staff. This was evaluated by means of interviews with plant maintenance and operational staff and by examination of plant personnel response to past experience related to PIV's.

The audit was extended to a Human Reliability Analysis (HRA), the scope of which was limited to an evaluation of human performance during a postulated ISLOCA event. A standardized list of variables, Performance Shaping Factors (PSF), which can have a positive or negative influence on the correct performance of an action was developed. This list was used to determine important human action/error. PSFs for each human action were then assessed in terms of potential positive or negative influence on performance. The licensee approach of system-based modeling technique and their PRA-driven HRA were also reviewed.

The audit, therefore, addresses the following six areas:

- o Identification of potential human errors and human actions, as related to ISLOCA events.
- o Evaluation of human performance shaping factors

- o Man-machine interface (MMI), including human performance (accessibility and environmental impact) during and after ISLOCA events.
- o Operating procedures: normal, abnormal, emergency, remote and local.
- o Training and operator knowledge.
- o Communication.

POSTULATION OF INTERFACING SYSTEM LOCA

Haddam Neck interfacing system boundaries are identified on the basis of potential ISLOCA pathways, at which the high and low pressure interfaces are separated by valves, operable either manually or remotely. Thus, the RCS isolation boundary of piping, heat exchangers, or RCS pump seals are excluded from consideration, and only those pathways that could be overpressurized by introducing RCS pressure, either due to inadvertent opening of valves or by failure of valves, were considered.

Assuming such overpressurization occurs, the interfacing low pressure systems may or may not be able to withstand the overpressurization depending upon the piping system, pathway, relief capacity, and magnitude of leak rate. The potential ISLOCA events at the Haddam Neck plant are therefore evaluated under three separate categories: ISLOCA without physical breaks, ISLOCA with small break, and ISLOCA with large break. Twelve potential ISLOCA events were identified for audit review.

The ISLOCA event without physical break or damage obviously implies that the physical integrity of the low pressure piping system would be maintained, and leakage through the PIVs would be well within the relief capacity of the system. In such an event, the pressure setpoint of the relief valve would be critical as well as the pathway of the contaminated RCS water. Another potential concern for this class of ISLOCA would be the cycling of the relief valve, resulting in a pressure-induced water hammer. The RHR system and relief valve, RH-RV-715, with a pathway leading to the Refueling Water Storage Tank (RWST) at Haddam Neck, may constitute a potential ISLOCA of this category.

When the leak rate from the RCS exceeds the relief capacity of a given line, the system may not be able to sustain the excess pressure, resulting in an ISLOCA with small or large break. Again, the audit activities focused not only on the size of the breaks but also the location of the potential breaks (inside or outside of containment) as well as the effect of the ISLOCA.

The audit was extended to activities beyond the design base accidents so as to evaluate the consequences of the postulated ISLOCA events and potential safety concerns. This included awareness of ISLOCA events by site personnel, ISLOCA preventive mechanisms or measures, programmatic weaknesses, and ISLOCA isolation and mitigation capability. The IST program was reviewed beyond the license conditions within the bounds of ISLOCA.

FINDINGS

The audit results were positive in those programs and activities required under the license conditions and prescribed in the regulatory requirements.

Also, the audit findings indicated that the plant programs designed to assure equipment and system integrity were adequate and that the plant staff exhibited excellent knowledge of plant operation. However, it was also clear that the plant staff members were not fully aware of ISLOCA events or their consequence. It appeared that this lack of awareness in ISLOCA events and consequences of the events might have contributed to the following negative findings as related to ISLOCA events:

POTENTIAL ISLOCA PRECURSOR EVENT

The "AS FOUND" leakage test results for two Pressure Isolation Valves (PIVs), SI-MOV-861B and SI-CV-862B, indicated that leakage across the two valves were 1.21 gpm and 50.9 gpm respectively on April 6, 1986. These two valves are located in the same discharge header, one of four, of the High Pressure Injection System (HPIS). The tests were part of the routine surveillance program. The acceptable leakage across these two valves (line leakage) is 1.0 gpm.

The licensee recognized the leakages and appeared to have performed corrective maintenance on these valves. This underscores the fact that existing surveillance requirements have a beneficial impact on the likelihood of ISLOCA. However, the significance of simultaneous leakage in two PIVs located in the same discharge header was not recognized as a breakdown in the pressure isolation capability of these PIVs and a potential ISLOCA precursor by the licensee. Furthermore, there was nothing in the test procedures which directed the test personnel or reviewers to evaluate the leakages of in-series PIVs for ISLOCA or ISLOCA precursor.

The valve leakages appeared to have occurred during shutdown conditions because the same two valves successfully passed surveillances performed on January 13, 1986. The plant was coasting down for the 13th refueling outage from January 4, 1986 through February 25, 1986, and was in mode 6 from February 25 to April 26, 1986. However, should the same leakage have occurred while the reactor was at power operations, detection of the leakage by the operators could have been delayed because there was no pressure instrumentation on the HPI discharge header nor were there any overpressurization alarms locally or in the control room. The header is equipped with a relief valve, SI-RV-870, with a capacity of 35 gpm at pressure lifting setpoint of 1500 psi. The discharge water from the relief valve is relieved to the Refueling Water Storage Tank (RWST), which is vented directly to atmosphere. The RWST is without radiation detection system or an alarm for possible RCS leakage flow.

However, according to the licensee's static analysis of the HPI discharge burst pressure, the HPI discharge piping system would withstand RCS pressure and temperature even though they were rated as only 1400 psi at 650° F and 1500 psi at 350° F. The licensee's PRA evaluation of ISLOCA indicated that the calculated core melt frequency for the HPI system is less than 10E-8 per year, which was less than 3% of overall ISLOCA contribution to the core melt frequency.

CORE DELUGE LINE:

Each of two Core Deluge lines has one MOV (SI-MOV-871A/B) and one check valve (SI-CV-872A/B), which constitute the pressure boundary interface between the high pressure RCS and the LPI/CD system. The inboard MOVs are welded to their corresponding check valve, and they are located on the reactor vessel head. Because of their proximity to the reactor, it is extremely difficult, if not impossible, to perform a leak test on these inboard MOVs. On the basis of the physical layout of the PIVs, these MOVs were exempted from normal ASME Section XI leak test requirements of category 'A' valves. In fact, under spurious Safety Injection Actuation Signals (SIAS) these MOVs could have been opening inadvertently and might not have closed completely.

The utility's own PRA study indicated that the core deluge line would be the highest contributor to the ISLOCA core melt frequency. On the upstream side of core deluge valves, SI-CV-872A/B, there is a hand-operated gate valve inside of containment. On the basis of common cause failure of safety injection systems, the licensee decided to change the actuator of the valve to a motor operated type, controllable from the Control Room, during the 1989 scheduled outage. If an ISLOCA were to occur due to the failure of the core deluge valves in series, this valve, SI-MOV-873, would be available to either minimize the loss of coolant inventory outside containment or to confine the loss of inventory to inside the containment. However, SI-MOV-873 could not be classified as a PIV because of the pressure rating of the piping. Also, the ability to close the valve under a high pressure gradient is questionable.

ALTERNATE LETDOWN LINE:

At the upstream side of the drain cooler common header (2" line), each RCS loop has one motor operated gate valve on each cold leg and one hand operated globe valve on each hot leg. These four (4) MOVs(DH-MOV-544, 534, 521, and 507) and four (4) manual valves (DH-V-539, 529, 516, and 502) on the four (4) RCS loops serve as the second pressure isolation boundary between the RCS and the Alternate Letdown Line. The other pressure interfacing PIVs are two parallel valves, DH-MOV-310 and DH-V-311. One of these two valves in conjunction with each one of the eight valves (4 MOVs and 4 manual valves) would constitute two pressure isolation boundaries.

However, these eight (8) valves were not treated as PIVs. The four MOVs are subjected to full exercise testing, but are not leak tested. The full exercise testing includes stroke time measurement and valve position indication. The four hand operated valves are categorized as passive valves by the licensee's valve IST program and are not subjected to any testing at all. Even though the licensee's IST program is in full compliance with the regulatory requirements, performing a leak test is the best way to ensure the integrity of these pressure isolation valves.

The Alternate Letdown system was designed to provide additional letdown capacity of RCS water-swelling during heatup/cooldown operations, and is normally isolated from RCS pressure during power operation. However, the cold leg MOV was routinely opened for a quarterly chemistry sampling, leaving DH-MOV-310 as the only pressure boundary valve.

EVALUATION OF HADDAM NECK ISLOCA

Several plant characteristics and features were observed that were specific to the Haddam Neck, and could influence the outcome of an ISLOCA event. Some of the positive attributes to the plant design include the Engineered Safety Features and their layouts. The HPI system can be partially, or in some ISLOCA scenarios, totally replaced by the charging system which is a safety-graded, large capacity, and high pressure system. The suction and discharge lines of the RHR system have two isolation MOVs in series, and the low pressure components of the RHR system (pumps, heat exchangers) are housed in a deep pit in the Primary Auxiliary Building (PAB). Furthermore, all of the PIVs, including that of the drain line, are located inside the containment building, and the low pressure piping of the interfacing systems are in general located in the pipe chase of the PAB.

- o Detection of ISLOCA

  Small leaks into the RHR system may not be readily detectable, and even if they are detected, it may be difficult to distinguish them from other non-ISLOCA leaks. Small leaks in the High Pressure Injection System are also difficult to detect due to lack of instrumentation, and may remain undetected for a relatively long time.

  Small or large breaks in the RHR pit or High and Low Pressure Injection pump areas would be detected readily by flood alarms, and area radiation monitors.

- o System/Pipe Integrity

  For the systems which dominate ISLOCA contribution to core melt frequency (RHR, LPI/CD, and HPI), it appears that the most likely break location is inside the containment. Furthermore, the most vital RHR equipment (i.e., pumps and heat exchangers) are located in a pit (sub-floor level in PAB), and a break in the RHR system outside of containment would be in the RHR pit area.

  The core deluge lines have a removable spool piece with flanges upstream of the check valves. This is the likely break location should an ISLOCA event occur.

  The low pressure lines which interface with high pressure piping systems are generally located in pipe trenches in the Primary Auxiliary Building (PAB), and are separated physically from other equipment and systems, minimizing potential interactions during ISLOCA events. However, the HPI and LPI pumps are housed in a common cubicle without physical separation. An ISLOCA event involving any one of the pumps may affect all others, incapacitating the remaining pumps under a common cause failure.

- o Isolation Capability

  All interfacing low pressure lines have remote isolation capability with MOVs. However, actual closure of these valves after an ISLOCA event is questionable due to uncertainty of their capability to close under a high pressure gradient. The planned installation of an additional MOV on the low pressure deluge line may provide additional assurance of this lines isolation.

The RHR suction (MOV-781) and discharge (MOV-803) valves are easily accessible and may be closed manually, if a containment entry is possible, during an ISLOCA event with a release outside of containment.

o Radiation Release

The PAB is relatively small with many pathways, directly to the outside environment. The structural walls at the second level have minimal pressure retention potential, and may give in to a minimal steam pressure from an ISLOCA. The building has a limited fire spray system, initiated by high area temperature, which may not exist during an event 'V'. The fire spray system may provide scrubbing action to limit the offsite release.

The RHR pit may provide a means to reduce the fission product release, if the break is in that general location. The accumulated water in the pit may provide beneficial scrubbing of fission products.

HUMAN PERFORMANCE: HUMAN FACTORS ENGINEERING

o Man-Machine Interface (MMI)

Information necessary to detect pressurization of the High Pressure Safety Injection system is not available in the control room. Also, a number of plant parameters, such as trend recording of Reactor Water Storage Tank (RWST) level, which might signal open relief valves in interfacing systems were not displayed in the control room.

o Emergency Operating Procedures

The "Response Not Obtained" (RNO) entries for an ISLOCA-related EOP step may not be feasible due to the inaccessiblity of the specified valves. Also, verbatim execution of the EOPs could initiate an ISLOCA event. The specific finding of this audit involves the operator incorrectly cycling open the core deluge valves following an inadvertent safety injection signal.

o Communications

Two aspects of the communications system could be improved to optimize operator performance during an ISLOCA event: formalization of procedures for transmitting information to and from the control room and provision of hand-held radio transceivers for use within containment.

- Training

  The set of existing training scenarios does not include an ISLOCA event, and the simulator is not capable of simulating most of the plant conditions typical of most ISLOCA events.

HUMAN PERFORMANCE: HUMAN RELIABLITY ANALYSIS (HRA)

A human reliability evaluation was performed to provide insight into the prevention/mitigation of ISLOCA in terms of human actions/errors. A number of valuable qualitative evaluations were accomplished. These include the identification of human actions/errors relevant to ISLOCA, and determination and assessment of relevant performance shaping factors (PSFs) for the identified actions/errors.

- Identification of Human Actions/Errors

  Specific actions/errors were identified and described in terms of system-based accident scenarios in which human actions played a major role. Four scenarios associated with low pressure injection/core deluge, three scenarios each associated with residual heat removal suction and injection, and four scenarios associated with high pressure injection were identified and defined in terms of human actions. Actions/errors identified in the scenarios were found to group into four important categories. These categories are: 1) operator-induced initiators, 2) operator actions as precursors, 3) maintenance actions as precursors, and 4) operator mitigation or aggravation; relevant to ISLOCA.

- Evaluation of Performance Shaping Factors

  PSF determinations and assessments are reported in a PSF ISLOCA scenario matrix format for ISLOCA events. This matrix allows identification and comparison of PSF against human action in the contexts of the ISLOCA scenarios in relation to assessments of positive or negative influences on human performance and reliability.

  Preliminary findings suggest that:

  - PSFs are generally positive for ISLOCA relevant actions that typically appear in other plant evolutions (not related to ISLOCA).
  - PSFs are generally negative for detection/diagnosis of ISLOCA situations, particularly for HPI.
  - The PSF "psychological stress" has potential negative influence on human reliability following a safety injection signal, or in off-normal situations.

- Personnel all appear to be sensitive to the importance of good communication. The PSF for communication is therefore generally positive. However, there are not enough physical lines of communication into the control room under certain complex ISLOCA emergency situations.

- This human reliability evaluation approach provides insight into prevention/mitigation of ISLOCA (e.g., identifying important actions and pinpointing PSFs which can reasonably be assumed to have a negative influence on reliability).

CONCLUSION

o The Audit team did not identify any regulatory issues.

o The NRC audit team's conclusion is that plant staff and operations were not fully aware of potential ISLOCA events nor understood their consequences. This unawareness could contribute to a lack of readiness in preventive, corrective, and mitigative measures of ISLOCA events.

o The NRC team concluded that within the scope of ISLOCA the licensee's programs, in general, are adequate but that there is some room for improvements in the maintenance program, EOPs, control room instrumentation, and training.

o The IST program is in full compliance with the regulatory requirements. However, there is no ISLOCA consideration incorporated into the program.

o It was concluded that the plant design is somewhat unique and atypical from an ISLOCA point of view due to its redundant HPI/Charging systems, type and location of the PIVs, and the RHR pit arrangements.

o The final list of 18 PSFs had been made for Haddam Neck. This list may vary from plant to plant. However, after three or four visits, it should be possible to determine a generic list.

This audit is part of a longer integrated effort to examine the ISLOCA issue. Further audits, as well as a balanced research program, will be conducted to make the final determination of the safety significance of ISLOCA.

DETAILS

1.0 Persons Contacted

Connecticut Yankee Atomic Power Company

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FOLDEN, J	REACTOR OPERATOR
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LEBARON, L	ELECTRICAL ENGINEER
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MILLES, B.	ELECTRICIAN
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*PERKINS, C.	NUSCO LICENSING COORDINATOR
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*SHEDLOSKY, T	SENIOR RESIDENT INSPECTOR, HADDAM NECK

* Denotes those present at the exit meeting on August 4, 1989

2.0 Audit Scope

2.1 Objective

The objectives were identification and collection of ISLOCA-related plant data, and "AS BUILT" and "AS FOUND" conditions. This information will provide a basis for subsequent evaluation of generic vulnerability of PWR ISLOCA events. The availability of plant features and human performance were evaluated to assure the plant's readiness and effectiveness to prevent or mitigate ISLOCA events.

2.2 Audit Methodology

The audit methodology employed a two step approach, the first of which was selecting the items to be audited. The second was the audit or review rationale.

2.2.1 Audit Items

The identification and selection of items which comprise the audit scope were based on the pathways of the RCS water from the high pressure primary system inside the containment to the other systems at lower pressures outside containment. Audit activities also included the identification of the failure modes in these leak pathways that could lead to release of the primary inventory to the outside environment.

On the basis of reviews of pertinent documents, including the Connecticut Yankee Probabilistic Safety Study (Appendix A) and the Haddam Neck System Descriptions (Appendix A), the following five systems were selected:

- RHR Systems
- High Pressure Injection System
- Low Pressure Injection and Core Deluge System
- Chemical and Volume Control System
- Alternate Letdown (Drain) Line

The high and low pressure systems were typically separated by a combination of two valves in series (two MOVs or one MOV and one check valve). The licensee's IST program identified twenty-two (22) PIVs. The audit team included an additional eight (8) valves for review. The isolation boundaries of piping, heat exchangers, and pump seals were not included in the audit on the basis of the licensee's PRA study results, and plant design and operational considerations.

Twelve (12) postulated ISLOCA events and potential leak pathways were identified involving the above five (5) systems. These postulated accident sequences provide insights to the potential event precursors and the mitigating system responses as well as human actions to successfully cope with the events. From this, specific system components (check valves or MOVs) which are required to operate; i.e., be available for system success were identified. The identified valves are:

a. RH-MOV-780 and RH-MOV-781 in the RHR Suction line:

The RHR pumps take a suction from the RCS loop 1 hot leg. The Inboard and outboard PIVs are RH-MOV-780 RH-MOV-781 respectively.

b. RH-MOV-804, 803 in RHR/LPI Injection line:

The Inboard and Outboard PIVs on the loop 2 cold leg injection line are RH-MOV-804 and 803 respectively.

c. SI-MOV-871A; 871B and SI-CV-872A, 872B in the Core Deluge Lines:

Two parallel trains with Inboard Isolation valves, SI-MOV-871A/B, and Outboard Isolation check valves, SI-CV-872A/B.

d. High Pressure Injection Valves:

Four (4) parallel trains, which each consist of one Inboard MOV (SI-MOV-861A/B/C/D) and one Outboard check valve (SI-CV-862A/B/C/D).

e. Valves in Letdown Line:

Three parallel flow control valves, LD-AOV-202, 203, 204, coupled with two PIVs, LD-MOV-200 and LD-AOV-230.

f. Drain Line Valves:

Two parallel Outboard PIVs, DH-MOV-310 and DH-V-311, and four (4) cold leg isolation valves, DH-MOV-507, 521, 534, and 544. Four (4) Inboard hot leg isolation valves, DH-V-502, 516, 529, and 539. The four (4) manual valves and four (4) cold leg isolation MOVs are not classified as PIVs.

2.2.2 Audit Rationale

The audit activities focused on licensee programs and activities that assure, or contribute to the availability of the equipment required to prevent or mitigate ISLOCA. Human performance and the ability of the station staff to effectively perform routine, reactive, or recovery actions to mitigate the ISLOCA were reviewed. A summary of the audit rationale is presented in figure 1.

Equipment availability is the ability of equipment to function when called upon to do so. Component and systems integrity must be maintained in order to respond to demand. This may be affected by aging, environment, operational time-cycles, surveillances and maintenance activities. Therefore, qualitative observations and reviews were made to evaluate the availability and integrity of equipment.

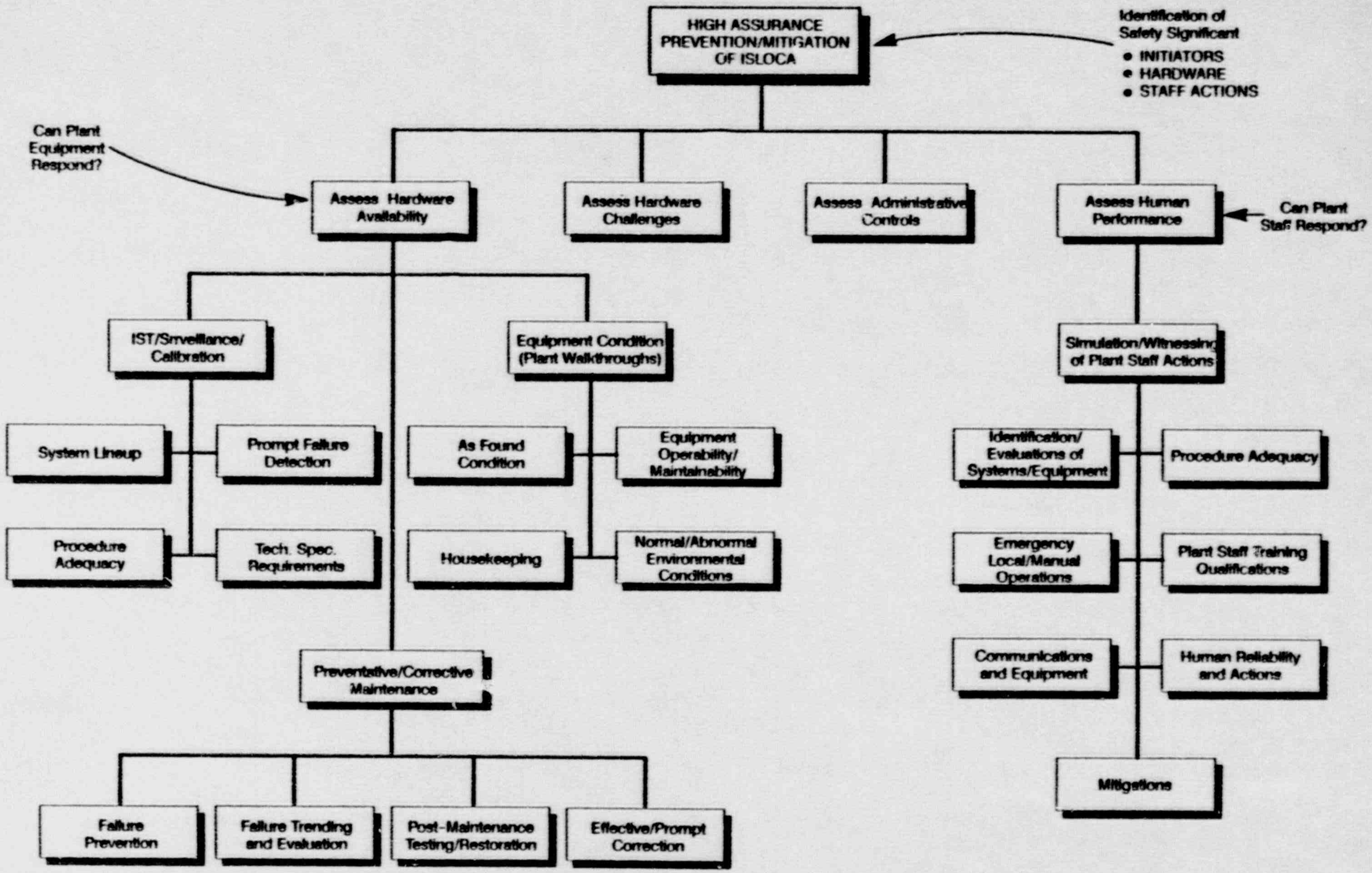

Figure 2.1 – Inspection Rationale of Interfacing System Loss of Coolant Accident.

Human actions and associated potential human errors were reviewed as well as factors which could affect human actions or potential human errors. Qualitative and subjective evaluations were included in the audit activity to determine Performance Shaping Factors (PSFs) which were relevant to the ISLOCA-related human actions.

2.2.2.1 Equipment Availability

The availability (operability) of plant equipment relevant to the ISLOCA events was evaluated. Also, those plant activities which contribute to, or verify equipment availability, relevant to the ISLOCA events were evaluated for effectiveness.

The following items were included in the audit to evaluate equipment availability:

- Preventive and corrective maintenance, and trending
- Return-to-service and post-maintenance testing
- Surveillance testing and records
- Accessibility of the equipment
- Alternate and emergency operability
- Visual inspections
- Component cooling, electrical supports, and ventilation
- Environmental qualification and fire protection
- I&C and calibration
- Procedures: normal, abnormal, alarm, and emergency
- Inservice Testing

The trending of equipment repairs included qualitative evaluation of maintenance and surveillance records, predictive maintenance activity, and engineering evaluation (i.e., burst pressure of pipes, calculated MOV torque values). Environmental effect evaluations were based on qualitative observations of adequacy of cooling, room humidity and temperature effects, and possible manual operations of MOVs inside containment during an ISLOCA situation. The equipment operability evaluation included demonstration of local operations, accessibility of the equipment and emergency lighting.

2.2.2.2 Human Performance

Operational readiness and the effective human performance were evaluated on the basis of the ability of the plant staff to detect, respond, and recover from the ISLOCA events. The factors affecting human performance were documented.

The following items were included in the audit to evaluate Human Performance:

- Adequacy and availability of procedures
- Completeness of information and procedures
- Demonstration of equipment operability, manual and remote
- "Hands-on" simulation of operations
- Test witnessing
- Operator interviews and system walk-through
- Training and qualification
- Operator alertness and human factors engineering
- Potential human errors and performance shaping factors
- Awareness of the ISLOCA events.
- Man-machine interactions and communication during ISLOCA

2.3 Other Aspects

The potential for a common mode failure and its root-cause was considered. Programmatic aspects of plant activities were evaluated in terms of effectiveness of the administrative controls and QA/QC, i.e., housekeeping, documentation, tagging, and onsite review processes of activities related to the ISLOCA events.

The licensee responses to NRC findings were also reviewed for technical adequacy and promptness. Utility initiatives were verified to assess their impacts on the quality of station activities, particularly those related to procedures and human performance.

The utility's initiatives on their Probabilistic Risk Assessment (PRA) study and the relevant Human Reliability Analysis (HRA) were reviewed for their roles in plant operations, design, and modifications. Particularly, the PRA/HRA modeling methodology as well as plant risk and the consequences of the ISLOCA events were discussed with the licensee's PRA/HRA specialists.

2.4 Records

It was desirable to review as many records as possible, particularly for corrective maintenance activities and trend evaluation of component repair and wear. However, the team was not able to review all the records for the past 20 years, and the audit activities were, in general, limited to the last five (5) years. The plant records reviewed include:

- Surveillance Test records
- Preventive maintenance records
- Corrective work orders
- Post-maintenance and functional test records
- Selected plant modification packages, including 10CFR50.59 review records
- Computer records (Production Maintenance Management System or PMMS)

Other supporting documents were reviewed to achieve the audit objectives. They include:

- o Plant Piping and Instrument Drawings (P&ID)
- o Connecticut Yankee Probabilistic Safety Study
- o System Descriptions
- o Plant "Q" listing
- o Applicable Licensee Event Reports (LERs)
- o FSAR
- o Station Administrative Control Procedures
- o Haddam Neck Technical Specifications

3.0 Summary of Findings

The major findings of the audit are summarized here for each of the interfacing systems. Numerous aspects of the various design features and operating practices as related to ISLOCA were examined in detail. The findings may be classified according to their specific roles in a potential overpressurization event. It is noted that physical location of equipment, operating practices, and/or different safety related requirements may lead to design arrangements that could result in an ISLOCA given particular circumstances. The licensee's present program, especially the probabilistic risk analysis area, recognizes these potential weaknesses, and proposals for design changes and improvements are being pursued and investigated.

3.1 High-Low Pressure Interfacing Systems

3.1.1 Residual Heat Removal System

1. Detection capability

   - o Small leaks into the Residual Heat Removal (RHR) system discharge piping through RH-MOV-803,804 or 780,781 are readily detectable if the discharge pressure indicator is periodically monitored. The discharge relief valve would relieve excess pressure to the Refueling Water Storage Tank (RWST), but its relief actuation opening is not indicated to control room operators and water level changes in the RWST may not be readily detectable. Also the cycling of the relief valve may damage the piping due to water hammer effects.

   - o Flood detectors would indicate any substantial water accumulation inside the RHR pump pit area. Radiation monitors are also installed inside the pit as well as in the Primary Auxiliary Building (PAB) to detect any radioactivity release.

2. System integrity

   - o The likely break locations of the low pressure RHR piping system is either inside the containment (pipe break) or in the RHR pump pit area (pump seals or heat exchangers).

- o RHR piping outside containment is located in the pipe chase of the PAB. Other vital Emergency Core Cooling System (ECCS) equipment is separated from the RHR piping.

3. Isolation capability

   - o Isolation valves (RH-MOV-781 and 804) inside the containment are easily accessible. Manual operation of these valves may be achieved under ISLOCA conditions.

4. Mitigation capability/consequences

   - o Potential break locations and water inventory loss are not addressed in the emergency operating procedures.

   - o Given a break due to an ISLOCA in the RHR pit area, potential "ad hoc" recirculation may be feasible using the accumulated water in the pump pit.

   - o The RHR pump pit may provide important scrubbing effects reducing the fission product release.

3.1.2 High Pressure Injection System

1. Detection capability

   - o Leak detection or monitoring instrumentation is not available for small leaks from the Reactor Coolant System (RCS) into the High Pressure Injection discharge line. The opening of the discharge relief valve and the consequent small release into the RWST cannot be directly detected. Indirect indications such as increased charging would be the primary means of detection.

   - o Flood detectors are installed in the HPI pump pit area.

2. System integrity

   - o The structural integrity of the HPI piping could remain intact upon a slow overpressurization event due to its design margin.

   - o The most likely pipe break locations appear to be inside the containment or in the pipe chase area of the PAB.

3. Isolation capability

   - o The Safety Injection (SI) stop valves are located inside the containment and even though access is relatively difficult, manual operation of the valves may be feasible.

4. Mitigation capability/consequences

- The physically separated charging system can be used as a backup system for the HPI system if the HPI system is not available or disabled.

- Physical separation between the HPI and LPI system components is not provided in the HPI/LPI pump area with potential spatial interactions.

3.1.3 Low Pressure Injection/Core Deluge System

1. Detection capability

- Small leaks are detectable using the RHR discharge pressure indicator.

- Flood detectors may indicate a substantial release in the LPI pump pit area.

- Radiation monitors would indicate any radioactivity release inside the PAB.

2. System integrity

- Most likely break locations are inside the containment, especially at the removable spool piece on each injection line.

- Potential break locations inside the PAB are likely at or near the LPI pump pit.

3. Isolation capability

- The core deluge valves, SI-MOV-871A and B are inaccessible for manual operations.

- The low pressure rated isolation valve, SI-MOV-873 is easily accessible and may be manually operated in an emergency.

4. Mitigation capability/consequences

- There is no physical separation provided between the LPI/HPI components in the pump pit with potential spatial effects given an ISLOCA in the general location.

- The core deluge isolation MOVs (SI-MOV-871A, B) are not leak tested to insure valve integrity.

- No pressure interlocks or administrative controls are available to prevent opening of these isolation valves.

3.1.4 Letdown System

1. Detection capability

   o Flow and temperature instrumentation is available to indicate overpressurization.

   o Volume Control Tank (VCT) level is monitored indicating a direct release through the letdown relief valves.

2. System integrity

   o The low pressure portion of the letdown piping is located in the pipe chase of the PAB. Any likely break would not directly affect any ECCS equipment.

3. Isolation capability

   o Two high pressure rated isolation valves are available with diverse power sources. One is motor operated and the other is air controlled with a back-up air supply bottle.

4. Mitigation capability/consequences

   o No major damage is expected to any of the ECCS equipment given an ISLOCA event in the letdown system.

3.1.5 Alternate Letdown System

1. Detection capability

   o Pressure and temperature indications are available to diagnose the conditions of the drain line before and after the drain isolation valve, DH-MOV-310.

2. System integrity

   o A large capacity relief valve is installed on the alternate letdown line inside the containment discharging to the sump.

   o The most likely break locations are inside containment at or near the relief valve.

   o Outside the containment, the potential break locations are in the pipe chase area or inside the RHR pump pit.

3. Isolation capability

   o Two low pressure rated containment isolation valves are available to isolate the drain line.

   o The drain isolation valve, DH-MOV-310, is accessible for manual operations if the break is outside the containment.

4. Mitigation capability/consequences

- o No major damage is expected to occur to the high pressure injection portion of the ECCS given an ISLOCA event in the alternate letdown system. However, the RHR pumps may be affected if the break location is at or near the Primary Drain Tank.

3.2 Valve and System Integrity

3.2.1 Maintenance Program

The team's audit of the maintenance program was concentrated in several areas as related to ISLOCA. These were: predictive maintenance, preventive maintenance, corrective maintenance, and post-maintenance testing.

- o It was found that the licensee presently has no predictive maintenance program in place. They are, however, developing this area and plan to initiate a predictive maintenance program in January 1990.

- o The licensee's preventive maintenance program appears adequate. Planning, scheduling, and coordination of maintenance activities are controlled well. Power operated pressure isolation valves receive periodic preventive maintenance as well as special MOVATS testing. Leakage data for these valves show a pattern of historical leak tight integrity which can be directly attributed to good preventive maintenance.

- o Corrective maintenance is also generally adequate, but may need improvement in one area. Safety injection check valve SI-CV-862B (discussed in Section 5.2 of this report as a contributor to a potential ISLOCA precursor in April 1986) has had significant internal rework performed in 1986 and 1987. The valve has failed numerous leak tests and was recommended to be replaced in 1987. However, as of the date of this audit the maintenance department had no plans to replace this valve. It is felt that improved oversight in the area of maintenance trending and replacement of problematic components is therefore warranted.

- o Post-maintenance testing was found to be adequately performed and administered. Retests/functional verification tests are required following maintenance on any Quality-Related system, structure, or component. This includes the pressure isolation valves of interest to this audit.

3.2.2 IST Program

Upon reviewing the Haddam Neck Valve IST Program, the NRC audit team identified the following findings:

- o Leak testing cannot be performed on core deluge valves SI-MOV-871A and 871B due to configuration and location. The licensee has been granted relief for this.
- o Eight (8) valves on the alternate letdown line are not being leak tested even though these valves can be considered as being the inboard pressure isolation valve.
- o Valve SI-MOV-873 is presently a hand operated valve but will be changed to motor actuation during the 1989 outage. This change in valve actuation will be very useful as a MOV to mitigate the consequences on an ISLOCA
- o All other appropriate valves have been identified as PIVs and are tested in accordance with ASME Section XI, Subsection IWV.

3.2.3 Surveillance Program

Upon reviewing the Haddam Neck Surveillance Program, the NRC audit team identified the following findings:

- o In general, the procedures were well written and technically adequate to perform their intended function.
- o All identified failures within the surveillance program were given immediate attention; i.e., corrective maintenance, post-maintenance functional testing.
- o The licensee is not routinely performing internal visual examination of any check valves in the HPI and core deluge header for degradation. The need for this is exemplified by the fact that valve SI-CV-872B was identified as having a worn hinge mechanism and under those circumstances would not function as a PIV.

3.2.4 I&C Program

The I&C Program, with respect to the implementation of controls for the motor operated pressure isolation valves of interest to this audit, was found to be adequate. The team reviewed the licensee's control of MOV torque switch bypasses, and settings for torque and position limit switches for the PIVs of interest. It was found that they were set correctly to allow proper valve operation with maximum differential pressures expected on these valves during both normal and abnormal events within design basis.

3.2.5 Administrative Controls

The licensee's overall program for administrative control of the maintenance and surveillance activities appears to be good within the audit scope of ISLOCA events. Some of the positive findings include:

- New equipment tagging program is being implemented.
- Procedure rewrite program is in place.
- Housekeeping is excellent.
- Temporary procedure changes and open item tracking system are well coordinated and implemented.

3.3 HUMAN PERFORMANCE

3.3.1 Human Factors Engineering

- Man-Machine Interface (MMI)

The relatively small size of the control room, in conjunction with system grouping and the use of mimics in panel layouts, provides control room operators with a satisfactory man-machine interface. The control room was quiet and conducive to effective voice communications. It was noted that there were very few "normally on" annunciators in the control room ("dark board") which contributed to a general impression of stability and control.

However, information necessary for early detection of an ISLOCA precursor condition in the HPI system was not available in the control room. There was no instrumentation in the HPI system for pressure, temperature or flow indication. As a result, the operators would have to rely on secondary indications, such as RWST level, to detect an ISLOCA precursor. This was not specifically addressed in the operating procedures. Furthermore, a number of plant parameter indications, such as trend recording of RWST level, which might signal the operator that an ISLOCA was evolving, are not displayed in the control room. These instruments are outside the control room. This may result in an increase in operator workload and additional coordination and communications requirements. Similarly, the absence of available panel space appeared to have prevented new equipment from being properly integrated with existing components.

o Operating and Maintenance Procedures

In general, the Haddam Neck operating procedures were well designed. They are complete, clearly written and well organized, and employ good layout and highlighting techniques, such as bold face and italics to emphasize important information, and boxed text for caution statements. The Emergency Operating Procedures (EOP) development and validation process includes iterative reviews by operations and training personnel, and formal feedback from simulation exercises. The EOP writer's guide was considered representative of the current state of practice in the industry.

The following conditions were identified:

a. For most ISLOCA scenarios, it is expected that the introduction of RCS pressure into the low pressure system will cause the associated relief valve to lift and vent to the RWST. An increase in RWST level may, therefore, be one of the first indications of an ISLOCA event. Despite the diagnostic value of RWST level, EOP E.0, did not direct the operator to check for a change in RWST level, nor did it explicitly refer to the possibility of an ISLOCA. Similarly, EOP ES-1.3, did not specify the significance of the absence of the expected water level in the containment sump as an indication of a potential ISLOCA. ECA 1.2,

b. With regard to the feasibility of required operator actions, in at least two instances, the "Response Not Obtained" (RNO) entries for a step directed the operator to manually operate valves that may not be accessible during an ISLOCA event. The RNO entries for EOP ECA 1.2, direct the operator to manually close valves that are located close to the reactor coolant loop RHR penetrations. Under ISLOCA conditions, manual operation of these valves may not be feasible.

c. In one instance, verbatim execution of the existing EOPS could exacerbate an evolving ISLOCA event. ECA 1.2, (Rev 1, 6/3/88), step 1.c, directs the operator to cycle the core deluge valves SI-MOV-871 A and B. Although the procedure directs the operator to monitor for reactor coolant system pressure change, the RNO for this step does not indicate the significance of a RCS pressure change as an indication of a potential ISLOCA precursor condition. As it is currently written, the procedure could inadvertently allow the operator to establish a ISLOCA precursor by cycling the MOV and leaving it in the open position.

d. The Haddem Neck EOPs contain numerous requirements for aux operators (AOs) to support control room operators during emergency conditions. For example, in an ISLOCA event involving SI initiation, failure of one or more of the RHR loop isolation valves, and an RWST alarm in the PAB, it is conceivable that four AOs would be required to perform the EOPs.

Several EOPs which are intended to accommodate ISLOCA are in the draft stage; however, these procedures were not in place at the time of the audit.

Electrical and mechanical maintenance procedures for the repair of RHR and SI MOV were well written using good human factors, practices and guidelines. They are arranged by logical action and verification steps that should enhance user comprehensibility and assure that the objective of the task is achieved. Caution and warning statements were highlighted, accurate, and did not contain action steps. Illustrations included in these procedures are accurate, clear and are generally located with applicable action steps. The vocabulary and abbreviations, acronyms and symbols used are consistent with plant standards. These are important to assure accurate ECCS valve maintenance, and should contribute to the prevention or mitigation of an ISLOCA.

o Communications

In general, the communications procedures and equipment are good. The control room is relatively small and quiet. This is conducive to effective voice communications. The public address and telephone systems were found adequate.

The telephone equipment is the primary mode of communication between control room operators and AO during a postulated ISLOCA event. Two aspects of the communications system which could be improved involve formalization of procedures for transmitting information to and from the control room and provision for hand-held radio transceivers for use within containment.

AOs reported that they do not routinely repeat back instructions received from control room operators, nor do they routinely write down component numbers and other difficult-to-recall information. However, the stated policy of the control room operators was to limit AOs actions to a maximum of two per instruction. Also, pertinent portions of selected procedures were available to AOs fcr sequences involving extended component lists.

In terms of providing hand-held radio transceivers for use within containment, it was determined that the location of telephones relative to probable operator locations within containment during postulated ISLOCA scenarios might require the operator to repeatedly transit between the valve location and the telephone which may be a distance of approximately 50 feet. This situation may unnecessarily contributes to operator workload and may increase the probability of error.

o Training

In general, the operator training program in place at Haddam Neck was determined to be adequate. The use of frequent and extensive simulation exercises utilizing a high-fidelity plant specific simulator and emphasis on problem solving skills were determined to adequately prepare operators to manage design basis events.

However, the simulator was not capable of simulating most ISLOCA events, and the plant models utilized by the simulator are incapable of simulating a failed ECCS check valve. In addition, there are currently no scenarios in the simulator training program which contain reactor coolant leaks which are large enough to require operators to initiate EOP ECA 1.2 "LOCA Outside Containment".

AOs receive a minimum of one day of simulator training per training cycle. This training emphasizes team coordination and communications between control room operators and AOs during simulated emergency conditions. This finding was considered to be a positive feature of Haddam Neck operational practices which should reduce the likelihood of communications problems between control room operators and AOs during a postulated ISLOCA event.

Feedback to the training staff on operating experience lessons learned was incorporated into applicable portions of the training program. However, information about plant behavior, such as ISLOCAs were not reflected in the program.

3.3.2 Identification of Human Errors

One focus of the audit was a human reliability evaluation relevant to ISLOCA. The overall goal of the human reliability evaluation is to provide insight into mitigation/prevention of ISLOCA in terms of human actions. Two human reliability evaluations of a qualitative nature were performed: 1) identification of important human actions relevant to ISLOCA, and 2) determination and assessment of performance shaping factors, that is, positive or negative influence on human performance for the identified actions (see Section 3.3.3).

The human reliability section for ISLOCA for the Haddam Neck PRA was reviewed. Human actions were described in terms of systems related ISLOCA scenarios. These scenarios were developed with focus on human actions to support human reliability evaluation. Four scenarios associated with low pressure injection/core deluge, three scenarios associated with residual heat removal suction, three associated with RHR injection, and four associated with high pressure injection were identified.

Human actions identified and described in the scenarios were grouped into four categories. These are:

1) Human initiators. (e.g., Operator opens a MOV, a leaky MOV/check valve already exists).

2) Human actions which are immediate precursors. (e.g., Operator improperly executes valve line up or fails to return valve to normal position following testing, in series with a second leaky/failed valve).

3) Human actions during repair which can compromise equipment. (e.g., Maintenance department installing wrong part or miswiring).

4) Human actions related to mitigation or aggravation. (e.g., Operator fails to detect ISLOCA or makes improper diagnosis).

3.3.3 Performance Shaping Factors

Performance shaping factors (PSFs) are variables which can have a positive or negative influence on the correct performance of an action by an individual or individuals in the context of human reliability. A standardized list of 22 PSFs, representing PSFs currently evaluated in HRA, were developed by the team prior to the Haddam Neck plant visit. This list was used to determine specifically which PSFs have potentially important impact on each human action identified and described in the ISLOCA human reliability sequences. PSFs determined for each identified human action were then assessed in terms of potential positive or negative influence on performance of action in regard to human reliability.

Data collection at the plant resulted in relevant information and specific patterns discernible in the matrix that was developed. Preliminary findings suggested included:

- PSFs are generally positive for ISLOCA relevant actions that typically appear in other plant evolutions.
- PSFs are generally negative for detection/diagnosis of ISLOCA situations, particularly for HPI.
- The PSF "psychological stress" has potential negative influence on human reliability following a safety injection signal.
- Personnel all appear to be sensitive to the importance of good communication. The PSF for communication is therefore generally positive. However, there may not be enough physical lines of communication into the control room, which may result in a negative influence.
- The human reliability evaluation approach followed provides insight into prevention/mitigation of ISLOCA (e.g., identifying important actions and pinpointing PSFs which can reasonably be assumed to have a negative influence on reliability).

3.4 AWARENESS OF ISLOCA

The audit findings, in general, indicated that the plant programs designed to assure equipment and system integrity were adequate and that the plant staff exhibited excellent knowledge of plant operation. However, it was also clear that the plant staff members were not fully aware of ISLOCA events or their consequence. It appeared that this lack of awareness in ISLOCA events and consequences of the events might have contributed to the following negative findings as related to ISLOCA events:

3.4.1 POTENTIAL ISLOCA PRECURSOR EVENT

The "AS FOUND" leakage test results for two Pressure Isolation Valves (PIVs), SI-MOV-861B and SI-CV-862B, indicated that leakage across the two valves were

1.21 gpm and 50.9 gpm, respectively on April 6, 1986. These two valves are located in the same discharge header, one of four, of the High Pressure Injection System (HPIS). The tests were part of the routine surveillance program. The acceptable leakage across these two valves (line leakage) is 1.0 gpm.

The licensee recognized the leakages and appeared to have performed corrective maintenance on these valves. This underscores the beneficial effect of existing surveillance requirements in controlling ISLOCA frequency. However, the significance of the simultaneous failure of two PIVs located in the same discharge header was not recognized as a breakdown in the pressure isolation capability of these PIVs and a potential ISLOCA precursor by the licensee. Furthermore, there was nothing in the test procedures which directed the test personnel or reviewers to evaluate the leakages of in-series PIVs for ISLOCA or ISLOCA presursor.

The valve leakages appeared to have occurred during shutdown conditions because the same two valves successfully passed surveillances performed on January 13, 1986. The plant was coasting down to the thirteenth refueling outage from January 4, 1986 through February 25, 1986, and was in mode 6 from February 25 to April 26, 1986. However, should the same leakge have occurred while the reactor was at power operations, detection of the leakage by the operators would probably have been delayed because there was no pressure instrumentation on the HPI discharge header nor were there any overpressurization alarms locally or in the control room.

3.4.2 CORE DELUGE LINE

Each of two Core Deluge Lines has one MOV (SI-MOV-871A/B) and one check valve (SI-CV-872A/B), which constitute the pressure boundary interface between the high pressure RCS and the LPI/CD system. The inboard MOVs are welded to their corresponding check valve, and they are located on the reactor vessel head. Because of their proximity to the reactor, it is extremely difficult, if not impossible, to perform a leak test on these inboard MOVs. On the basis of the physical layout of the PIVs, these MOVs were exempted from normal ASME Section XI leak test requirements of category 'A' valves. In fact, under spurious Safety Injection Actuation Signals (SIAS), these MOVs could have been opening inadvertently and might not have closed completely.

3.4.3 ALTERNATE LETDOWN LINE:

At the upstream side of the drain cooler common header (2 inch line), each RCS loop has one motor operated gate valve on each cold leg and one hand operated globe valve on each hot leg. These four (4) MOVs (DH-MOV-544, 534, 521, and 507) and four (4) manual valves (DH-V-539, 529, 516, and 502) on the four (4) RCS loops serve as the second pressure isolation boundary between the RCS and the Alternate Letdown Line. The other pressure interfacing PIVs are two parallel valves, DH-MOV-310 and DH-V-311. One of these two valves in conunction with each one of the eight valves (4 MOVs and 4 manual valves) would constitute two pressure isolation boundaries.

However, these eight (8) valves were not treated as PIVs. The four MOVs are subjected to full exercise testing, but are not leak tested. The full exercise testing includes stroke time measurement and valve position indication. Furthermore this cold leg MOV was routinely opened for a quarterly chemistry sampling, leaving DH-MOV-310 as the only PIV in the line.

4.0 High-Low Pressure Interfacing Systems

4.1 Residual Heat Removal System

4.1.1 System Description

The Residual Heat Removal (RHR) system is part of the Emergency Core Cooling System (ECCS) providing Heat removal capability during normal cooldown or shutdown operations and in the recirculation phase of a postulated Loss-of-Coolant accident (LOCA). The RHR system consists of two pumps, two heat exchangers, isolation valves on the suction and discharge lines and associated piping, instrumentation and control (see Figure 4.1).

For normal cooldown or shutdown operations, the RHR pumps take suction from the loop 1 hot leg through two motor operated valves exchangers to the loop 2 cold leg through RH-MOV-803 and 804. During normal plant operations RHR is aligned for standby ECCS operations. All isolation valves (RH-MOV-780, 781, 803, and 804) are closed and service water is aligned to the secondary side of the RHR heat exchangers.

In the injection phase of ECCS operations, the RHR system is passive and injection flow is provided by the Low Pressure Injection pumps. In the recirculation phase, the RHR system is used to remove residual heat from the reactor core via the RHR heat exchangers and return the recirculated water to the Reactor Coolant System (RCS) through the core deluge valves, SI-MOV-871A and 871B.

The piping of the RHR system is designed for 600 psig and 650 F, which is lower than the normal operating pressure of the RCS at 2000 psig. The high pressure RCS, designed for 2500 psig and 650 F, is separated from the low pressure RHR piping by two isolation valves on each interfacing line, i.e., RH-MOV-780 and 781 on the suction line and RH-MOV-803 and 804 on the discharge line. The low pressure piping is equipped with a large capacity relief valve on the RHR pumps discharge side relieving excess pressure to the Refueling Water Storage Tank (RWST).

4.1.2 System Interface Configuration

The high pressure RCS loops are separated from the low pressure RHR system using two isolation valves in series on each interfacing line. The inboard isolation valve, RH-MOV-780 and the outboard MOV, RH-MOV-781 are used on the RHR pumps suction line to isolate the loop 1 hot leg from the RHR system. On the discharge side of the RHR pumps, the inboard isolation valve RH-MOV-803 and the outboard RH-MOV-804 isolate the RHR heat exchangers from loop 2 of the RCS. The interface configuration between the high and low pressure piping is shown in Figure 4.2.

The discharge header of the RHR pumps has a pressure indicator in the control room. All four isolation valves are provided with valve position indicating lights on the control board. Red lights indicate open position, green closed and when both are lighted the valve is traveling from one position to the other. The lights are powered from a separate circuit, independent of the motor operator, allowing position indications even when the motor operator circuit breakers are racked out.

The valves are also interlocked with RCS pressure preventing their opening unless the RCS pressure is less than RHR operating pressure.

In addition to the pressure interlock, the outboard valves, RH-MOV-781 and 804, are provided with the following features to prevent inadvertent overpressurization of the RHR system:

- o The valve control switch on the Main Control Board (MCB) is key interlocked, i.e., a key must be inserted into it before the switch may be turned.

- o Disconnect switches, located in the control room, are provided between the circuit breakers on Motor Control Center 5 (MCC-5) and the motor operators of the valves. The disconnects are normally locked open and administratively controlled.

- o The closing of the disconnects alarms on the MCB.

- o An alarm is also actuated on the MCB when any of the outboard valves (RH-MOV-781 or 804) are traveling open.

The above listed features were closely inspected. Control room operators were questioned about the detailed operations of these isolation valves. It was noted, by visual examination of MCC-5, that all the circuit breakers for the isolation valves were racked in. This may allow the inadvertent operation of the inboard valves if the pressure interlock is bypassed and the pressure drop through the valves is less than the design delta p for opening (the motor operators of these valves are undersized, i.e., cannot open the valve if the delta p across the valve is larger than 500 psig). Even relatively small leakage rates, well below the allowable test limits, could equalize the pressure on both sides of the inboard valves over a period of time. This would essentially eliminate the ISLOCA preventive mechanism associated with the underdesign of the motor operator.

The various preventive mechanisms as well as those which could actually lead to the overpressurization of the low pressure systems (with other concurrent events) are listed in Table 4.1.

4.1.3 RHR system integrity

The RHR system design includes certain instrumentation providing essential information to the operators about the condition of the system and operating parameters. Some of these instruments may be used by the operators to recognize and assess an over-pressurization event. The primary indication available on the MCB is the RHR pumps discharge pressure. Additional instrumentation such as flood detectors and radiation area monitors in the RHR pit area and radiation monitors in other locations of the Primary Auxiliary Building (PAB) may also be used to help in diagnosing and determining the potential effects or affected equipment due to an Interfacing ISLOCA.

A relatively small ISLOCA event through the suction or discharge line may be readily detectable using the discharge pressure indicator. The relative large capacity relief valve (RH-RV-715) would relieve pressures in excess of 50 psig which is well beyond the normal pressure of the RHR system during the standby mode (only residual pressure). However, other concurrent small leak events from the RCS may mask this type of ISLOCA, if the discharge pressure is not regularly checked and monitored (the set point for the relief valve is 500 psig).

A system functional test was identified, SUR 5.1-4 "Core Cooling Systems Hot Operational Test", that includes a procedure on how to relieve residual pressure of the RHR discharge line after an operational test. This procedure may inadvertently be used to relieve the system pressure due to a small ISLOCA event. Primary coolant would be released to the primary sample system through a small high pressure rated instrument line.

The RHR system isolation valves (RH-MOV-780, 781, 803 and 804) are regularly leak tested after refueling outages during normal ascent to full power operation (SUR 5.1-1, "Hydrostatic Test").

During a walk-down of the RHR system the following observations were noted with regard to a potential ISLOCA event:

o Isolation valves, RH-MOV-781 and 804 are located inside the containment in easily accessible areas. During an ISLOCA event an attempt may be made to manually operate these valves. The valve operators are underdesigned to prevent valve movement if the delta p across the valve is larger than 500 psig. The low pressure piping arrangement between the valves and the containment wall penetration contains a number of vertical and horizontal pipe segments with elbow pieces. These are judged as the most likely locations of a postulated pipe break due to an ISLOCA event. In this case, the ISLOCA is inside the containment and the water inventory accumulates in the containment sump. However, recirculation capability may still be impaired, since the RHR suction piping from the hot leg and the containment sump is connected.

- o The portion of the system outside the containment is located in a deep pit in the PAB. The RHR pumps and heat exchangers are located in their own separated cubicle inside the pit area. The pipe runs between the RHR pit and the containment wall penetration are located in the pipe chase, separated from all other vital ECCS equipment. The RHR pit contains a flood and radiation area monitor, that would indicate leakage from the RHR system. Any accumulation of a small leak can be pumped out by a submersible sump pump preventing large scale flooding of any cubicle. Potential damage to the RHR pump motors from humidity or other adverse conditions may therefore be minimized. The most likely break locations are the RHR pump seals, the heat exchangers and various valves at the boundary of the piping system.

4.1.4 ISLOCA scenarios

In order to categorize the possible consequences of an ISLOCA event, a number of different accident scenarios were examined during the walk-downs. Operators were questioned about operational responses and possible actions available to them. The relative effects of an ISLOCA occurring either on the suction or discharge lines (RH-MOV-780, 781 or RH-MOV-803, 804 respectively) are similar due to the RHR pump recirculation line, that allows pressure equalization between the suction and discharge lines.

The following discussion is valid for an ISLOCA event occurring in either the RHR suction or discharge lines.

1. ISLOCA without physical break or damage

   In this event, the leak rate from the RCS is within the relief capacity of the RHR system relief valve, RH-RV-715. Primary coolant would be discharged directly to the RWST. The diagnosis of this event may be difficult since the pressure would oscillate in the discharge line, and the RWST level would not increase dramatically. The potential problem of this event is the cycling of the relief valve, which could induce water hammer and damage the RHR system piping.

2. ISLOCA-small break

   The leak rate from the RCS may exceed the relief valve capacity overpressurizing the RHR system. The potential small break locations are:

   o the RHR pump seals

   o the RHR heat exchangers

   o closed valves at the pipe boundaries such as;
   containment sump check valves RH-CV-783 and 808A;
   RWST tie check valve RH-CV-784;
   LPI pump discharge check valve SI-CV-103;
   HPI pump recirculation MOVs SI-MOV-901 and 902.

o the RHR system piping either inside or outside of the containment

The RHR pump seals appear to be the most likely break location. These pumps are located in the pit area separated from each other by a concrete wall. A flood and radiation area monitor would indicate any leakage in this area. The damage to one pump may be contained and subsequent effects on the other pump may be minimal. Similar conclusions may be drawn for a break involving the RHR heat exchangers. The availability of the vital ECCS equipment would most likely be unaffected, especially the HPI pumps, which are located at a separate location .

If the break occurs at the various boundary valves, the RWST and the containment sump may be isolated from the RHR system using the isolation MOVs, RH-MOV-21,SI-MOV-901 and 902. It seems likely that the HPI portion of the ECCS remains intact and the environment in the PAB would not preclude its operation. In addition to the HPI pumps, the charging pumps can also inject water to the RCS from the RWST. An attempt can also be made to isolate the small break ISLOCA, if one of the isolation valves are operable and the break is outside the containment. However, the delta p across the valves may be such as to prevent their operation.

The RCS may be depressurized using the steam generators with the auxiliary feedwater system. By depressurizing the RCS in this way, RWST water inventory loss can be minimized.

Make up capability to the RWST is relatively small, but the procedure for this is well established and available to the operator.

3. ISLOCA-large break

The potential break locations are similar to the small break ISLOCA with the exception of the RHR pump seals. Regardless of the location of the break, inside or outside containment, the recirculation capability would in all likelihood be lost since the break would affect the RHR suction and/or discharge pipes. The LPI system would automatically start up injecting cold water from the RWST. The loss of RHR discharge pipe integrity would directly prevent core cooling, since the injected water may be lost through the break.

Recognition of potential RWST water inventory loss, as related to the location of the large break, may be a difficult task. The operators are not sensitized to this potential coupling effect in the ISLOCA procedures (ECA-1.2, "LOCA Outside Containment").

If the break is outside the containment, the line can be isolated using the isolation valves, if these are available. However, this attempt may be accomplished only after the RCS is depressurized, which is expected to occur in a few minutes after the break.

The environment in the PAB is expected to be very severe, since neither the RHR pit nor the pipe trenches are leak tight. The primary coolant release through the break in the PAB would pressurize the building, and any attempt to enter the PAB would be difficult. There is a very likely potential that all ECCS equipment, such as the HPI, LPI and charging pumps would be severely affected by the environmental conditions in the PAB.

In a large break ISLOCA, the recirculation capability may directly be lost, the LPI could be rendered ineffective (even if the environmental effects would allow the operation of this system) and the RWST water inventory could be depleted rather quickly.

These scenarios were discussed with the operators and training personnel. The availability of the various equipment, actual locations and potential interactions were verified during the audit process.

4.2 High Pressure Injection System

4.2.1 System Description

The High Pressure Safety Injection system is part of the ECCS and provides high head injection following a LOCA. After accident initiation the HPI pumps start and inject water from the RWST into the RCS. The HPI system may also be used in the recirculation phase to provide high head recirculation. The simplified schematic of the HPI system is shown in Figure 4.3

The HPI system is normally aligned for standby operation with the pumps secured and the HPI loop stop valves, SI-MOV-861A, B, C and D, are shut. The suction of the pumps are connected to the RWST. A minimum flow recirculation path is provided back to the RWST allowing to test the pump and the various components during normal plant operations.

There are four injection lines, each with two isolation valves, an MOV, SI-MOV-861A,B,C and D and a check valve, SI-CV-862A,B,C and D. Each line is equipped with a recirculation line for sampling and flow testing. These lines have additional manual isolation valves, SI-V-863A, B, C, and D.

When safety injection is initiated, the four loop isolation valves (MOV-861A through D) are automatically opened and the HPI pumps are started and borated water is injected into the RCS cold legs. The HPI flow is manually stopped by the operator during the transfer to the low pressure recirculation mode. If high head recirculation is required, then the RHR pump discharge is aligned to the HPI pump suction and the water is recirculated through the HPI lines.

The HPI system piping is designed to 1400 psi at 650°F (1500 psi at 350°F) and each injection line is isolated from the high pressure RCS by the loop isolation valve and a check valve in series. A 1500 psi relief valve, SI-RV-870, of the common discharge header protects the HPI system from overpressurization (maximum flow capacity 35 gpm).

4.2.2 System Interface Configuration

The interface between the RCS and the HPI system consists of two high pressure isolation valves on each injection line. The inboard isolation valve is the SI loop stop valve, (SI-MOV-861A, B, C, and D). It has a remote controlled motor operator. Outboard isolation is provided by self actuating check valves (SI-CV-862A, B, C, and D).

The four loop stop valves are controlled by three-position control switches from the MCB and the valve positions are indicated by lights (red and green for open and closed positions respectively, both are lighted when the valve is traveling or throttled). The circuit breakers of the MOVs on MCC-5 are normally energized allowing the opening of these valves on SI signal. This is a prerequisite for safety injection purposes, but from ISLOCA point of view, the valves may inadvertently open either by a spurious SI signal or through human error, by bypassing the control logic.

The outboard self actuating check valves and the inboard isolation MOVs are leak tested regularly at every refueling outage. In addition, a monthly operational test is also performed for the check valves, using SUR 5.1-4 "Core Cooling Systems Hot Operational Test". During the test, the check valves are unseated and a minimum flow is established. At the completion of the test, the check valves are reseated by gravity. The periodic performance of this operational flow test could significantly increase the reseat failure probability of these isolation check valves.

4.2.3 System integrity

The HPI system discharge piping has no remote or local pressure indication. The small capacity relief valve is also not indicated. Hence, small ISLOCA events may go unnoticed. In this case, the primary coolant would be discharged to the RWST, which has a very large volume, making the diagnosis relatively difficult. The operator would rely on the RCS volumetric balance measurements using the Volume Control Tank level indication and other parameters. Other instrumentation available to the operator is the flood detector at the HPI pump pit area and the radiation monitors in the PAB.

The HPI system was visually inspected and the following observations were made:

- Long sections of the low pressure HPI piping are located inside the containment with a number of horizontal and vertical sections and elbows. It is highly probable that given an ISLOCA event the pipe would be breached inside the containment.

- o The HPI piping, between the containment penetration and the HPI pump pit inside the PAB, is located in the pipe chase separated from other ECCS equipment. However, the pipe chase itself is not leak tight and a major release, even if it is inside the pipe chase area, could overpressurize the PAB.

- o The HPI pumps are located in an open pit area together with the LPI pumps. No spatial or physical barrier is provided between the different safety trains and components of either the HPI or LPI pumps. This physical arrangement could potentially lead to the loss of both systems due to an ISLOCA event occurring on the discharge line of either the HPI or LPI pumps.

- o The pump pit is monitored by flood detectors for water accumulation. Given an ISLOCA event and a pipe break in this location, the release is probably in the form of steam that may not condense in the open area to set off the flood detectors.

- o The charging system is physically separated from the HPI pump pit and could serve as an alternate means of high pressure injection to the RCS. The two charging pumps are in separate open cubicles and as such may be affected by large releases inside the PAB either from the HPI or RHR system.

Static calculations were made on the pressure integrity of the HPI piping regarding an overpressurization event. The calculations have indicated that the HPI piping may withstand a slow, static ISLOCA event (see Section 5.7). This essentially assumes that the failure of the interface boundary is such that only a relatively small leak is developed, which slowly overpressurizes the HPI discharge header. Other failure modes, such as a sudden opening, may also have to be considered since these impose additional dynamic loadings on the piping system.

4.2.4 ISLOCA scenarios

An overpressurization event on the HPI lines may lead to different consequences depending on pipe structural integrity, relief valve capacity, isolation boundary failure mode and potential operator actions. Based on these considerations, the following group of scenarios were investigated in detail during the audit:

1. ISLOCA, small leak no structural damage

   In this case, the leak rate through the isolation boundary is relatively small and the HPI discharge relief valve, SI-RV-870 (max., flow rate 35 GPM) would open discharging to the RWST. The operator would not diagnose the problem directly, but rather only through the RCS imbalance indications using the level monitors of the Volume Control Tank (VCT).

2. ISLOCA, small break

The leak rate through the isolation boundary may exceed the relief valve capacity leading to an overpressurization event that may damage either the piping sections or the HPI pump discharge check valves (SI-CV-856A,B). If the break is inside the containment, the charging pumps could provide high head injection and even recirculation could be accomplished by aligning the RHR discharge to the charging pump suction through opening RH-MOV-33A and B. This operation may also be attempted even if the break is in the PAB, since the charging pumps are located in an area that is physically separated from the HPI pump pit.

If the break is in the HPI pump pit area, the concern is the spatial effects on the HPI and LPI pumps. However, the operation of the LPI pumps are not required for high head injection or recirculation and the function of the HPI pumps may be partially replaced by the charging pumps.

3. ISLOCA, large break

The size of the high pressure rated section of the HPI line to each of the cold legs is 3". Therefore, the failure of the isolation boundary cannot lead to very large breaks because the pipe size limits the outflow of the primary coolant from the RCS. If the LOCA leads to very rapid depressurization, the operation of the LPI is required in the injection phase. If the break is inside the containment, both the LPI and RHR system is available to cool the core. An outside break, especially near the HPI pump pit, may impair the operation of the LPI pumps. However, the RHR system would still be intact and would be available for injection of cooling water from the RWST. Water inventory would not directly be lost. The inventory would be lost only through the break in the PAB achieving once-through cooling of the reactor core. The charging pumps could also be used to limit the loss of the water inventory, but their operation may not be assured upon a large release in the PAB.

4.3 Low Pressure Injection/Core Deluge System

4.3.1. System Description

The Low Pressure Safety Injection system is part of the ECCS and is designed to automatically deliver borated water to the reactor vessel following a LOCA. The major components and flowpaths of the LPI are shown in Figure 4.5.

Immediately following an accident initiation the LPI pumps start injecting water from the RWST into the RCS. After the injection phase, the water is collected in the containment sump and is recirculated using the RHR system. The LPI system is used only in the injection phase, and essentially corresponds to the accumulator systems of other pressurized water reactors.

The LPI system is normally aligned for standby operation with the pumps secured and the core deluge isolation valves, SI-MOV-871A and B shut. The suction of the pumps is connected to the RWST. A minimum flow recirculation path to the RWST is maintained by the throttled recirculation valves. The discharge of the LPI pumps is also connected to the RHR discharge piping to supply the Core Deluge Containment Spray, Charcoal Filter Spray and RCS loop 2.

During injection phase, the core deluge isolation valves, SI-MOV-871A and B open and borated water is injected through the deluge piping which penetrates the top of the reactor vessel. Once initiated, the LPI flow continues until terminated by the operator during the switch-over to the recirculation mode of operation.

The piping of the LPI system is designed for 600 psi at 650° F up to the core deluge check valves, SI-CV-872A and B. The system is separated from the high pressure RCS system by two normally closed isolation valves on each interfacing line (SI-MOV-871A, SI-CV-872A and 871B, 872B). The LPI discharge line shares a common header with the RHR system which is equipped with a relief valve, RH-RV-751. Any overpressurization of the common header either from the LPI or RHR interfacing lines would be relieved through this valve to the RWST.

4.3.2. System Configuration

The interface between the low pressure LPI piping (600 psi) and the RCS (normal operating pressure 2000 psig) is comprised of two isolation valves on each injection pathway as shown on Figure 4.6. The core deluge valves, MOV-871A and B, are the inboard isolation motor operated valves. The outboard isolation valves, CV-872A and B, are self actuated check valves.

The core deluge valves are controlled by three position control switches on the MCB with spring return to the auto position. These valves are normally closed and may automatically open on a safety injection signal. The motor operators are powered from MCC-5 with the circuit breakers racked in, so that the valves may be operational under normal conditions.

This particular feature is not desirable from an ISLOCA point of view, since any inadvertent Safety Injection (SI) signal may open the valve. This may initiate an overpressurization event, if the downstream check valve, CV-872, is in the failed position or malfunctioned. The various mechanisms relating to an ISLOCA event regarding the isolation components in the LPI system are listed in Table 4.1.

The isolation MOVs have position indications on the MCB (red and green for open or closed position). An additional indication is also available to determine if control power is available for the motor control circuit.

The discharge header common with the RHR discharge line has a pressure indicator (PI-602 on the MCB) that may indicate an overpressurization condition. However, the source of the overpressurization is not limited to the core deluge lines, but could also be the RHR discharge line through RH-MOV-803 and 804. The actual identification of the particular leak pathway is difficult from the operational point of view.

The core deluge line has one low pressure rated additional isolation valve, SI-MOV-873 (will become operational in the next outage). This valve may also be used to isolate an ISLOCA event. The use would depend on the flow and pressure conditions through the valve, as indicated by the extent of the accident.

4.3.3 System Integrity

Instrumentation is available for the operator to diagnose an ISLOCA event through the core deluge lines. Common RHR/LPI discharge header pressure indicator is located on the MCB. Relatively small leaks are not readily detectable, since the relief valve (RH-RV-715) would relieve excess pressure to the RWST.

Additional instrumentation is available in the LPI pump area to detect leakages in the pump pit (flood detector, may not be reliable upon steam release in the area) and radiation monitors are also located in the PAB indicating and alarming any radioactivity release inside the building.

The isolation check valves are regularly leak tested. However, the inboard MOVs are only stroke tested to insure operability. The plant ISLOCA probabilistic risk analysis has indicated that the highest risk of an ISLOCA event is potential overpressurization events of the core deluge lines.

The following observations were noted regarding the general arrangement and physical separation of the LPI system:

o The core deluge isolation valves, MOV-871A,B, are physically located in a relatively inaccessible area that would prevent the manual operation of these valves during an emergency. It is unlikely that the valves could be operated, even if the containment environment would allow it. However, the location of the low pressure rated isolation valve, SI-MOV-873, is between the primary shield and outer containment wall with relatively easy access and manual operation of this valve could easily be accomplished.

o Each core deluge injection line has a removable flanged spool piece. It is most likely that given an overpressurization event the LPI pipe integrity would be breached at this location inside containment. In addition, the piping arrangement would most likely insure that any pipe break (other than the spool piece) would occur inside the containment. However, the rcirculation capability would still be impaired due to the common header arrangement with the RHR discharge line.

o Most of the piping sections outside the containment are located in the pipe chase of the PAB. Any failure in this location would not directly affect other vital equipment, unless the PAB becomes overpressurized. The LPI pumps are located in an open pump pit together with the HPI pumps and valves. Physical separation is not provided between any of the components of this vital ECCS equipment. If an ISLOCA event occurs at the discharge check valve of the LPI pumps (SI-CV-103), all of this ECCS equipment may be affected impairing the core cooling capability of the plant.

4.3.4. ISLOCA Scenarios

The effects of an ISLOCA event through the core deluge lines are essentially identical to the scenarios discussed in the RHR system description due to the common header arrangement. It is important to note that not only the LPI system is affected by the core deluge line ISLOCA, but most likely the RHR system would also be damaged including the RHR pump seals and heat exchangers The potential break locations are:

o the RHR pump seals

o the RHR heat exchangers

o closed valves at the pipe boundaries such as; containment sump check valves RH-CV-783 and 808A RWST, tie check valve RH-CV-784, LPI pump discharge check valve SI-CV-103, HPI pump recirculation MOVs SI-MOV-901 and 902

o the LPI/RHR system piping either inside or outside of the containment

Depending on the severity of the damage both the injection and/or recirculation capability would be lost, leading to core damage. In addition to the scenarios described in section 4.1.4, the following potential scenario was discussed in great details with the operators:

1. ISLOCA-small/large break at LPI pumps

In this case, the break or damage occurs at or near the HPI discharge line check valve, SI-CV-103. This postulated event is important in that the HPI pumps are located adjacent to the LPI pumps and would in all certainty be damaged by the release through the break. The flood detector in the pump pit and the radiation monitors in the PAB would alert the operator to the abnormal event.

Even if HPI capability is lost, the operator could use the charging pumps to inject water into the RCS. The charging pumps are located in separate pump cubicles. However, these cubicles are open to the main level of the PAB and some potential environmental effect may be expected from a relatively large release in the LPI pump area.

A potentially more serious event would be a large break at the LPI pump discharge lines. The LPI system would automatically start injecting water from the RWST and through the break the water inventory would rapidly be depleted. The interaction between the location of the break and water inventory is not recognized by any procedure and the operators are not fully aware of the potential situation.

Flow instrumentation on the LPI discharge line is not available and the operators rely on pump motor current indications. This is an indirect method and probably unreliable during a large break ISLOCA event in the PAB due to environmental effects.

4.4. Chemical and Volume Control System

4.4.1 System Description

One of the primary purposes of the Chemical and Volume Control System (CVCS) is to maintain the proper water inventory in the RCS. It continuously provides purification of the primary coolant to reduce corrosion and radioactive products.

The letdown system, a subsystem of the CVCS, normally supplies a continuous bleed of reactor coolant to the chemical and volume control equipment of the CVCS. The normal letdown taps off the loop 1 cold leg through a "crud trap" that is reduced to a 3" line. It flows through a number of regenerative heat exchangers cooling the letdown flow.

There are two isolation valves located in the letdown line. A motor operated valve, CD-MOV-200 upstream of the heat exchangers and the letdown containment isolation valve, CD-AOV-23 which is air operated.

Outside the containment the letdown flow is directed through pressure reducing orifices and flow control valves (CD-FCV-202,203,204). The letdown line is designed to low pressure (600 psi at 650 F) downstream of the flow control valves. The pressure isolation function is essentially performed by the letdown orifices which are passive devices. Two large capacity relief valves (CD-RV-205,252) are installed downstream of the flow control valves discharging to the Volume Control Tank (VCT) and protects the low pressure segment of the CVCS against overpressurization.

A simplified schematic, depicting the letdown pathway, is shown in Figure 4.7.

4.4.2 System Configuration

The interface between the low pressure portion of the letdown line and the RCS is unusual, since the isolation is performed by passive orifices and orifice control valves. In addition, two normally open isolation valves are also installed on the high pressure portion of the letdown system.

The orifice valves (FCV-202,203,204) are air operated and fail closed upon loss of air supply. Three position (OPEN-AUTO-CLOSE) control switches are used to control valve position which can be throttled to control coolant flow. Three indicating lights are installed on the MCB (red and green for open and closed and white for auto). The valves automatically isolate the letdown flow upon receiving either high containment pressure (HCP) or low pressurizer level signals.

Inside the containment two high pressure rated isolation valves are available on the letdown line. The letdown containment isolation valve, LD-AOV-230 is an air operated valve which fails closed upon loss of air or DC power. A back up air bottle, located inside the containment would automatically supply air pressure, if the normal supply is lost. The valve will also automatically close on HCP to isolate the containment and letdown flow. The valve can be controlled from the MCB with a special override feature.

The other high pressure isolation valve, LD-MOV-200, is a motor operated valve that closes automatically upon actuation of the SI signal. The valve can be controlled from the MCB and position indicating lights are available above the control switch.

4.4.3 System Integrity

A postulated ISLOCA event in a letdown system is somewhat different than those occurring in normally closed interfacing low pressure systems such as the RHR and LPI. The letdown system is normally isolated by passive pressure reducing devices and orifice flow control valves which allow a constant flow of primary coolant into the CVCS. Overpressurization may occur due to the failure of these pressure and flow reducing elements or through other failure mechanisms such as sudden flow surge or failure of the letdown flow path in the CVCS.

The primary indications available to the operator are the letdown flow and temperature monitors on the MCB that shows the system conditions downstream of the non-regenerative heat exchanger. Upon overpressurization, the relief valve may open discharging excess flow to the VCT, where the level is also monitored and displayed on the MCB. If the ISLOCA event results in a small break, the VCT level monitor is used to diagnose the small LOCA conditions.

The letdown system, piping, valves and other components were visually checked during the audit "walkthrough" process. The containment isolation valve is located in the outer annulus area of the containment. However, its location is such that manual operation of the valve is unlikely. The letdown piping, orifices and orifice flow control valves are all located in the pipe chase area of the PAB separated from vital ECCS equipment.

The piping, just before the non-regenerative heat exchanger, is exposed, but only non-essential auxiliary components would be affected by any potential pipe break in this location. The VCT is located on the top floor of the PAB in an enclosed cubicle. The tank is relatively large and overpressurization from the opening of the letdown relief valves is unlikely. The top floor of the PAB contains mostly non-essential air conditioning, filtering and other equipments which would not be required during an ISLOCA event.

4.4.4 ISLOCA scenarios

An overpressurization event in the letdown line could be caused by either the unlikely failure of the pressure reducing devices, the sudden opening of a normally closed orifice flow control valve (flow surge) or a blocking of the letdown path by the failure of the low pressure control valves of the CVCS. The relief valves would open to relieve the excess pressure, but pipe or component damage may still occur due to dynamic forces.

1. ISLOCA without any damage

   In this case, the relief valve opens and the excess pressure and flow is relieved to the VCT. The capacity of the VCT is relatively large, but an excess primary coolant flow could overpressurize the tank. No major ECCS components are affected. Isolation is assured by the two high pressure rated isolation valves or by the orifice flow control valves which automatically close upon receiving either the SI or HCP signals. The diagnosis of the event is judged to be relatively easy and proper operator actions may be accomplished in a timely manner.

2. ISLOCA-small break

   The overpressurization of the letdown line may result in a pipe break in the PAB. The release would be detected by the radiation monitors in the building stack enabling the operator to diagnose this event. Isolation capability remains intact and is automatically initiated upon the decrease in the RCS pressure (SI signal). Manual actuation is also possible using the control switches on the MCB.

   Manual closure of any of the isolation valves inside or outside the containment is less likely given their particular locations. The orifice flow control valves are located in the pipe chase area and any release in this environment would prevent operator entry. The damage inside the PAB is expected to be minimal without a major effect on the operation of the vital ECCS equipment.

4.5 Alternate Letdown/Drain System

4.5.1 System Description

The alternate letdown flowpath is used when the normal letdown flowpath is not available and the reactor is shutdown. Each RC loop has two drain flowpaths, one attached to the cold leg and isolated by an MOV (DH-MOV-507,521,534,544). The other pathway may allow the draining of the RCS through the hot leg utilizing two manual valves on each line (DH-V-502,502A,516,516A,529,529A and 539,539A respectively). The drain flow from each loop is combined into a header and is directed to either the Primary Drain Tank (PDT) or VCT. The drain header flow is controlled by a motor operated valve, DH-MOV-310. The valve is not qualified for post accident conditions and as such the line is not used during normal operations. The system layout is shown schematically on Figure 4.8.

The alternate letdown line is also used to regularly take samples from the RCS during normal operations by opening the isolation MOV on the cold leg drain line ("sample procedure to be added"). During this operation the flow control valve, DH-MOV-310, is closed and a small sample is directed to the sampling system.

The design pressure of the piping downstream of the drain flow control valve MOV-310 is 150 psi at 500 F which is substantially lower than the RCS operating pressure at 2000 psig. The system is protected against overpressurization by a large capacity relief valve, DH-RV-1847 which is installed on the drain header downstream of MOV-310 inside the containment. The relief valve discharges onto the floor and any outflow is eventually collected in the containment sump.

4.5.2 System Interface Configuration

The high and low pressure piping of the alternate letdown system is separated by either two MOVs in series on the cold leg drain lines (DH-MOV-310 and DH-MOV-507,521,534 or 544) or by one MOV and two manual globe valves on the hot leg drain lines (DH-MOV-310 and DH-V-502, 502A, 516, 516A, 529,529A or 539,539A).

The flow control valve MOV-310 may be opened and throttled by a control switch from the MCB and is provided by two position indicating lights. During normal operation, the valve is not allowed to be open and its motor control circuit breaker on MCC-5 is racked out with a lockout feature. This was visually inspected and verified during the audit. The valve is regularly leak tested and its integrity is verified.

The loop drain valves (MOV-507,521,534,544) have also control switches on the MCB with position indicator lights. The valves are presently not leak tested, but operability stroke test is performed in each refueling outage. These valves are always operational during normal operation and periodically are opened for RCS sampling.

Outside the containment, the drain line is equipped with two air operated containment isolation valves which close automatically upon the HCP signal or the loss of air supply. These valves are low pressure rated and essentially are not part of the interface boundary.

4.5.3 System Integrity

The successful diagnosis of an ISLOCA through the alternate letdown line may depend on the severity of the event. Two primary indications are available on the MCB. A pressure indicator, (PI-108), is located at the upstream of the flow control valve MOV-310 which is the space between the two isolation boundaries for all the drain lines. A temperature indicator is also installed downstream of MOV-310 measuring the drain header temperature.

However, the cold leg MOV was routinely opened for a quarterly chemistry sampling purpose and the MOV-310 is the only PIV separating the low and high pressure piping systems.

The low temperature readings of the downstream drain header, has given the operators a fairly reliable diagnosis of the system conditions. A small leak would be diagnosed by the lower pressure indicator readings (PI-108) with increasing temperature indications. In addition to these direct indications, the level in PDT is alarmed when it reaches a certain high level.

However, there are other incoming lines in the PDT especially from the RC pump seal returns and the operators would routinely concentrate on that system before the alternate letdown would be investigated.

If the relief valve, DH-RV-1847 opens, the discharged primary coolant would eventually be collected in the containment sump and an increasing sump level indication is also available on the MCB to help diagnose the event.

The location of MOV-310 was verified along with the relief valve. Based on the piping arrangement, the drain header seems to be the most likely location of a postulated pipe break given an overpressurization event. The relief valve may also cycle open-closed that could enduce potentially damaging water hammer effects.

Outside the containment, the alternate letdown piping is in the pipe chase and the PDT is located inside the RHR pump pit. The potential effects of a pipe break on any of these locations would be localized and with the possible exception of the RHR pumps no major ECCS equipment would be directly affected. The RHR pumps are separated from the PDT by concrete walls that would limit direct damage to the pumps or motors.

4.5.4 ISLOCA Scenarios

An ISLOCA event in the alternate letdown line would be essentially a small LOCA event, since the maximum release is limited by the drain line sizes from either the cold or hot leg that is 2".

1. ISLOCA without damage

The drain header relief valve would open releasing primary coolant into the containment. The concern is the cycling of the relief valve that could damage the low pressure piping. The damage would be in all probability localized to the drain header inside the containment. No effect is expected on any of the vital ECCS equipment.

2. ISLOCA, small break

The break location is, with high probability, inside the containment without any effect on HPI capability. If the break is outside, it may occur in the pipe chase area or inside the RHR pump pit. The radiation area monitor inside the pit area or in the PAB would alarm helping to diagnose the event by the operators. The operation of the RHR pumps could be affected by the release inside the pit. However, the coolant loss is limited and the use of the HPI or charging system is sufficient to mitigate the small LOCA.

TABLE 4.1

DESIGN FEATURES TO PREVENT OVERPRESSURIZATION

OF LOW PRESSURE SYSTEMS

SYSTEM	VALVE CONFIGURATION	OVERPRESSURIZATION PREVENTIVE MECHANISMS	MECHANISM THAT COULD LEAD TO OVERPRESSURIZATION
RHR SUCTION	INBOARD ISOLATION RH-MOV-780	- RCS PRESSURE PERMISSIVE INTERLOCK FOR OPENING - CANNOT OPEN AGAINST LARGE DELTA p	- MOTOR OPERATOR CIRCUIT BREAKER CLOSED DURING NORMAL OPERATION
	OUTBOARD ISOLATION	- RCS PRESSURE PERMISSIVE INTERLOCK FOR OPENING - ELECTRIC POWER DISCONNECT SWITCH IN CONTROL ROOM, KEY LOCKED - KEY OPERATED CONTROL SWITCH ON CONTROL BOARD - RHR DISCHARGE PRESSURE INDICATION - VALVE TRAVELING ALARM DISCONNECT CLOSED ALARM - CANNOT OPEN AGAINST LARGE DELTA p	
RHR/LPI INJECTION	INBOARD ISOLATION RH-MOV-804	- RCS PRESSURE PERMISSIVE INTERLOCK FOR OPENING - CANNOT OPEN AGAINST LARGE DELTA p	- MOTOR OPERATOR CIRCUIT BREAKER CLOSED DURING NORMAL OPERATION

TABLE 4.1

(CONTINUED)

SYSTEM	VALVE CONFIGURATION	OVERPRESSURIZATION PREVENTIVE MECHANISMS	MECHANISM THAT COULD LEAD TO OVERPRESSURIZATION
	OUTBOARD ISOLATION RH-MOV-803	- RCS PRESSURE PERMISSIVE INTERLOCK FOR OPENING - ELECTRIC POWER DISCONNECT SWITCH IN CONTROL ROOM, KEY LOCKED - KEY OPERATED CONTROL SWITCH ON CONTROL BOARD - RHR DISCHARGE PRESSURE INDICATION - VALVE TRAVELING ALARM DISCONNECT CLOSED ALARM - CANNOT OPEN AGAINST LARGE DELTA p	
CODE DELUGE MOV LINES A/B	INBOARD ISOLATION SI-MOV-871A/B		- SAFETY INJECTION ACTUATION SIGNAL - NO LEAK TESTING PERFORMED
	OUTBOARD ISOLATION SI-CV-872A/B CHECK VALVE	- RHR DISCHARGE PRESSURE INDICATION	
HPI LINES A/B/C/D	INBOARD ISOLATION MOV SI-MOV-861A/B/C/D		- SAFETY INJECTION ACTUATION SIGNAL
	OUTBOARD ISOLATION CHECK VALVE SI-CV-862A/B/C/D		- NO DISCHARGE PRESSURE INDICATION
DRAIN LINES	OUTBOARD ISOLATION MOV DH-MOV-310	- MOTOR OPERATOR CIRCUIT BREAKER RACKET OUT WITH KEY CONTROL - PRESSURE INDICATION	

TABLE 4.1

(CONTINUED)

SYSTEM	VALVE CONFIGURATION	OVERPRESSURIZATION PREVENTIVE MECHANISMS	MECHANISM THAT COULD LEAD TO OVERPRESSURIZATION
	OUTBOARD ISOLATION BYPASS VALVE DH-V-311		
	INBOARD LOOP ISOLATION, COLD LEG DH-MOV-507 521 534 544		- PERIODICALLY OPEN FOR RCS SAMPLING - MOTOR OPERATOR CIRCUIT BREAKER CLOSED
	INBOARD LOOP ISOLATION, HOT LEG DH-V-502, 502A 516, 516A 529, 529A 539, 539A		
LETDOWN	ISOLATION VALVE LD-MOV-200	- SAFETY INJECTION (SI) SIGNAL ACTUATES VALVE	
	ISOLATION VALVE LD-TV 230	- HIGH CONTAINMENT PRESSURE (HCP) SIGNAL ACTUATES VALVE (SI GENERATES HCP SIGNAL AS WELL)	
	ISOLATION VALVE LD-AOV-202 203 204	- HIGH CONTAINMENT PRESSURE SIGNAL VALVE	
		- FLOW LIMITING MECHANICAL DESIGN	

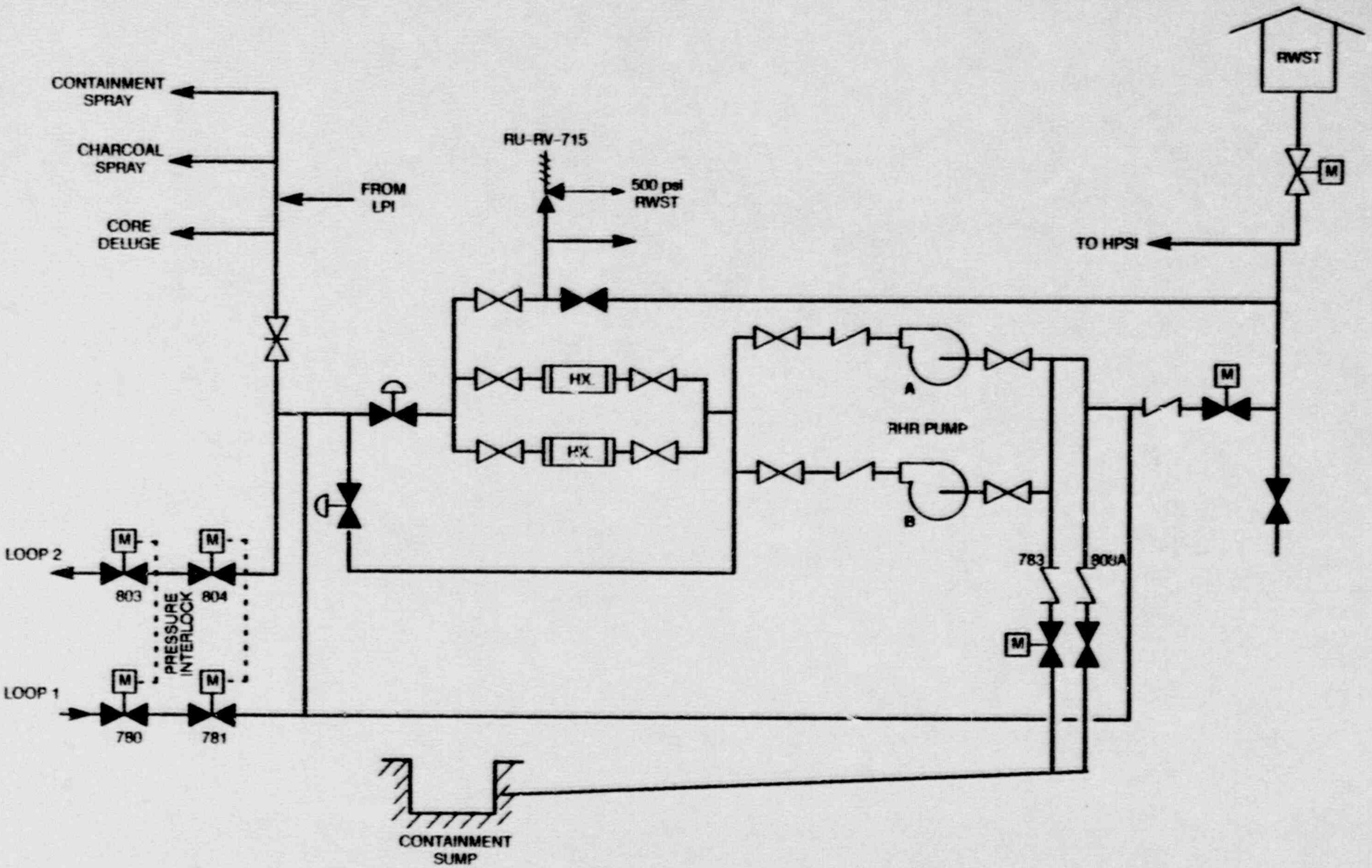

Figure 4.1 – RESIDUAL HEAT REMOVAL SYSTEM

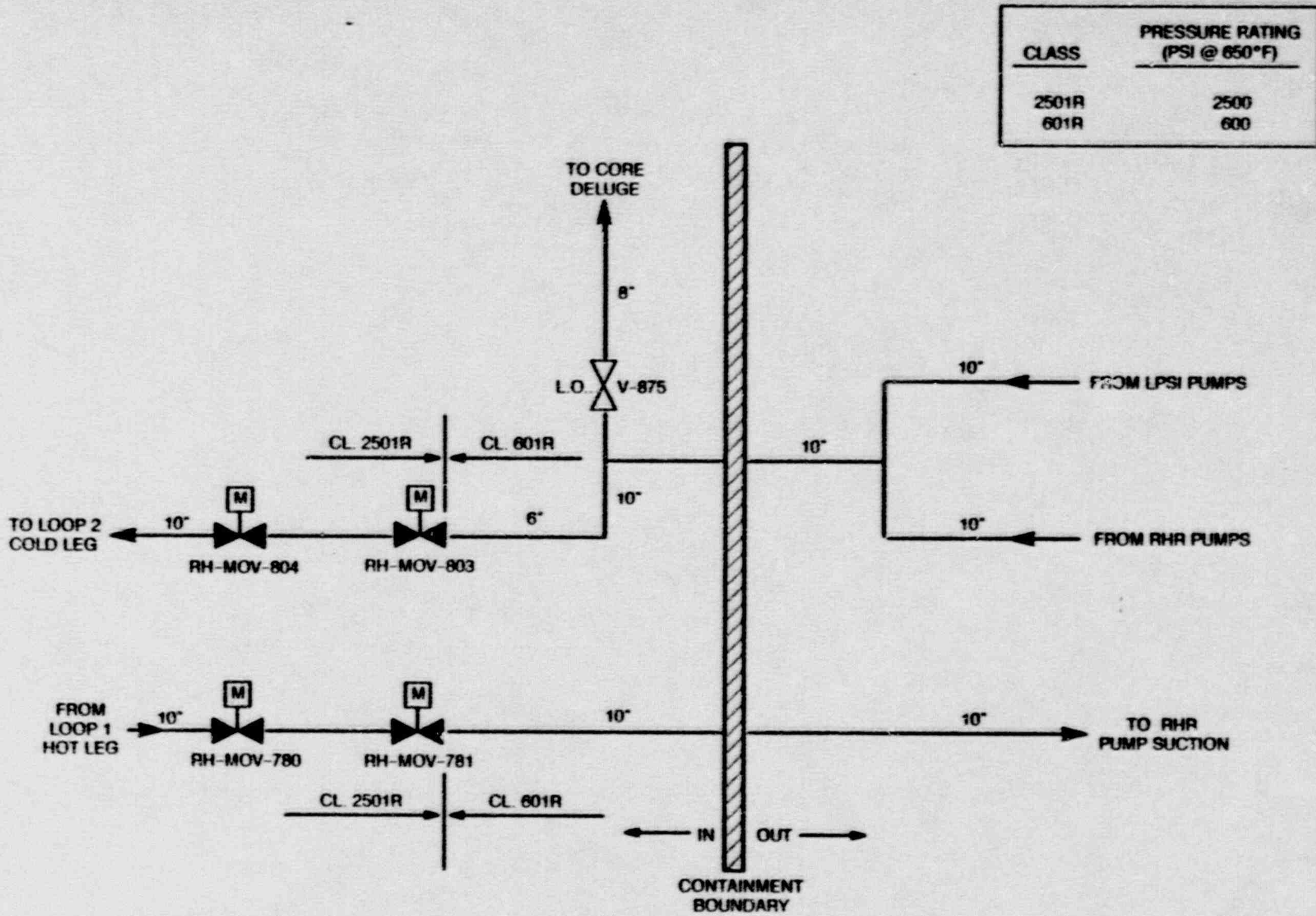

Figure 4.2 – HIGH/LOW PRESSURE INTERFACE – RHR SYSTEM

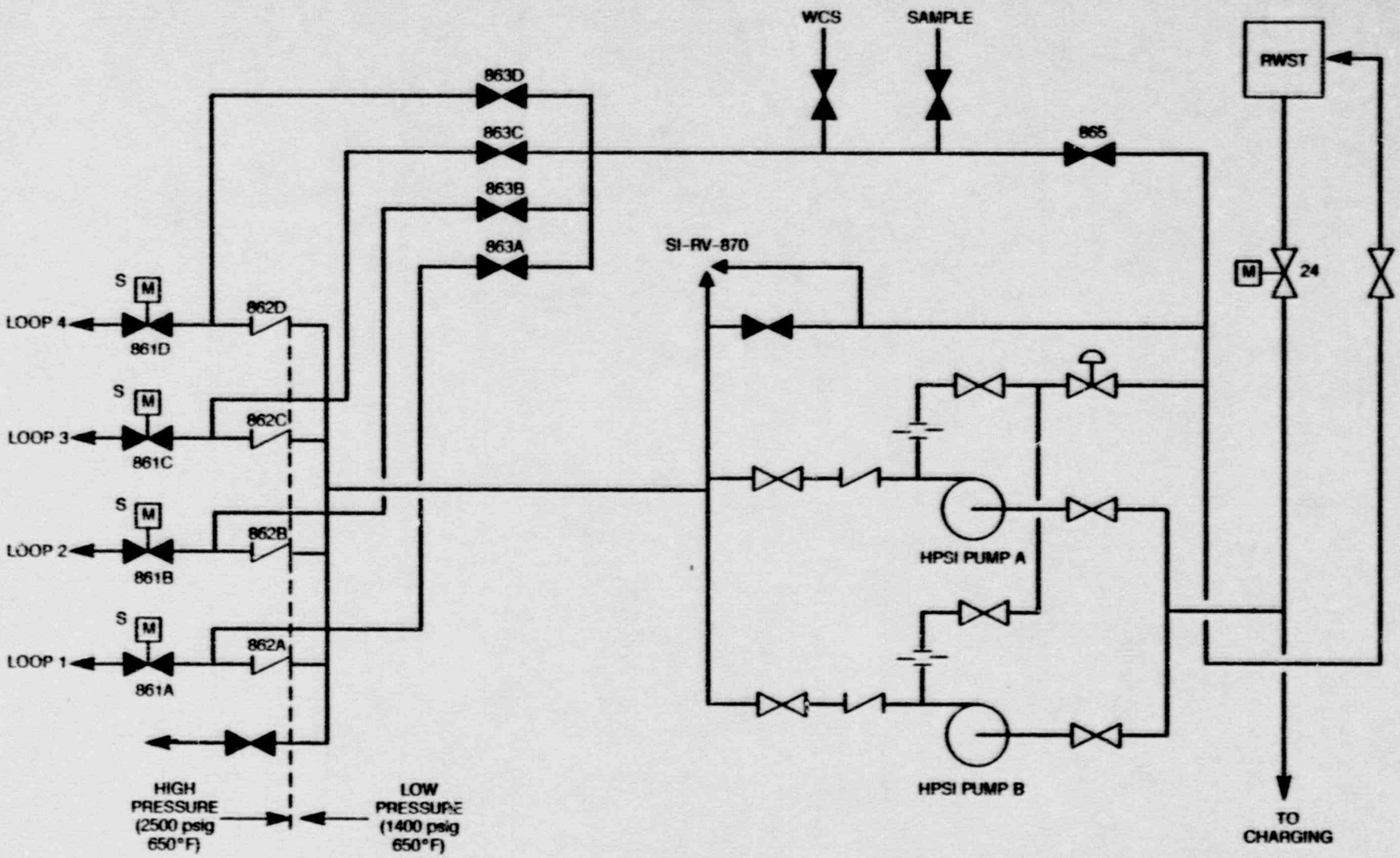

Figure 4.3 – HIGH PRESSURE SAFETY INJECTION

CLASS	PRESSURE RATING (PSI @ 650°F)
2501R	2500
1501R	1400

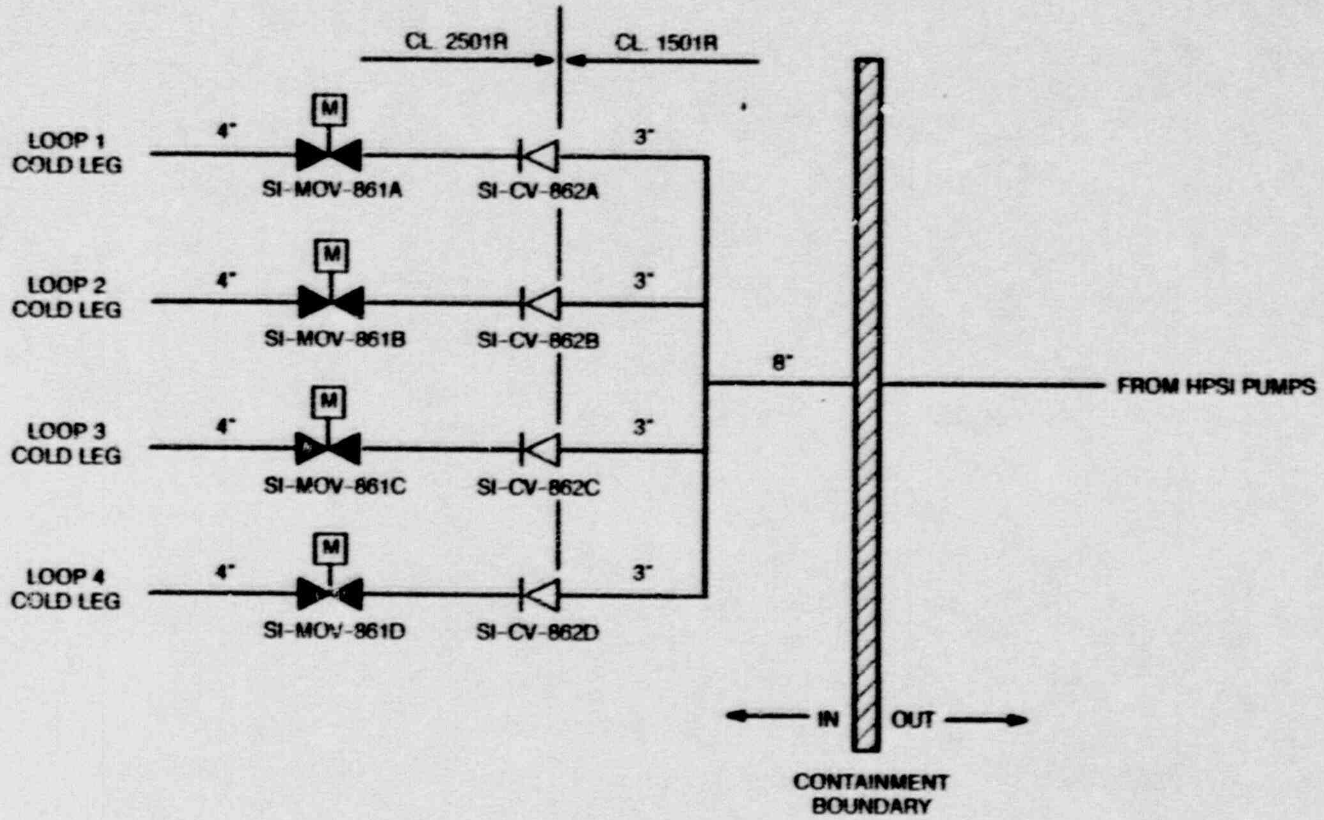

Figure 4.4a – HIGH/LOW PRESSURE INTERFACE – HPSI SYSTEM

CLASS	PRESSURE RATING (PSI)
2501R	2500 @ 650°F
601R	2100 @ 650°F
152	150 @ 550°F

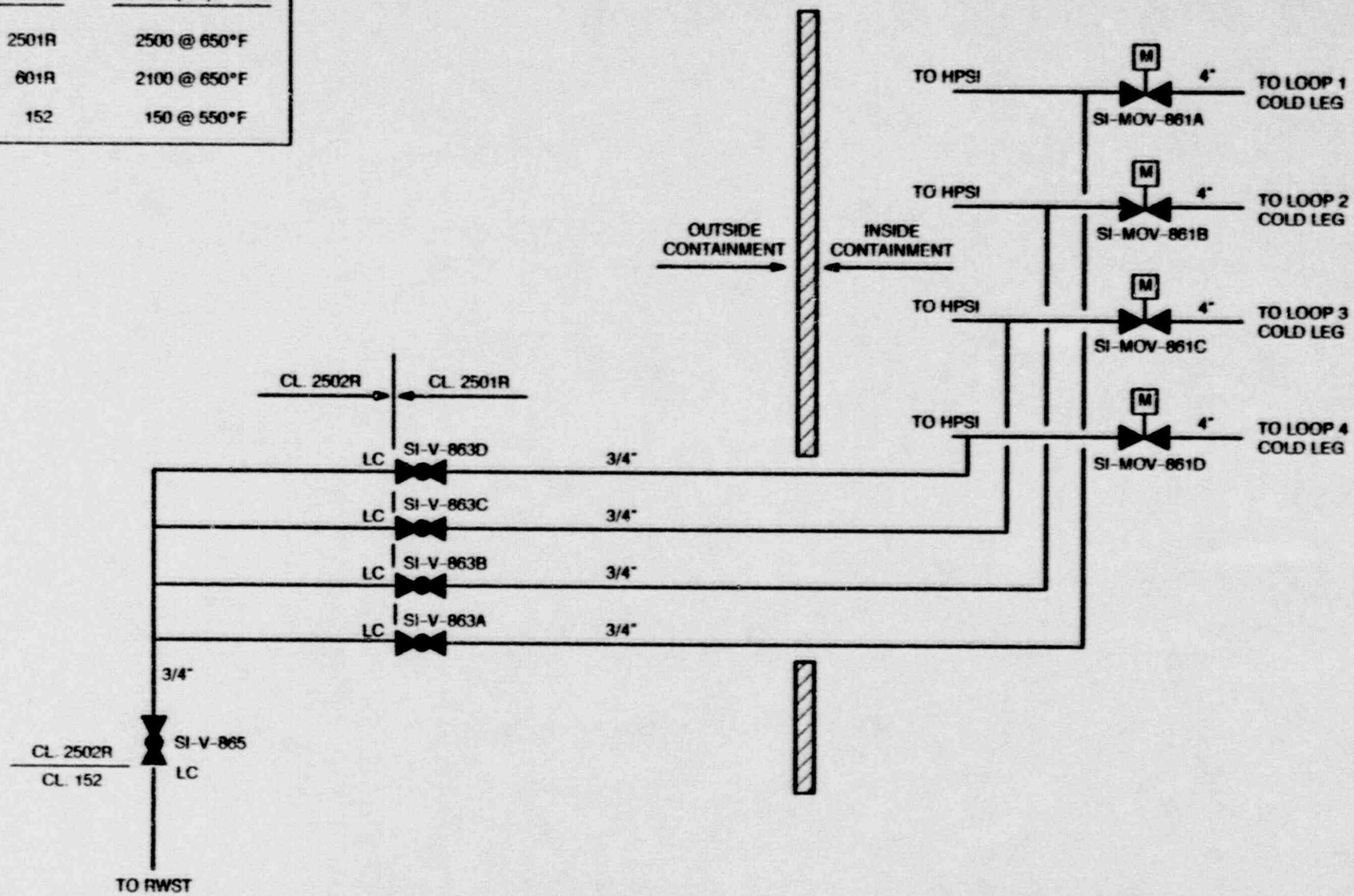

Figure 4.4b – HIGH/LOW PRESSURE INTERFACE – SI RECIRCULATION LINES

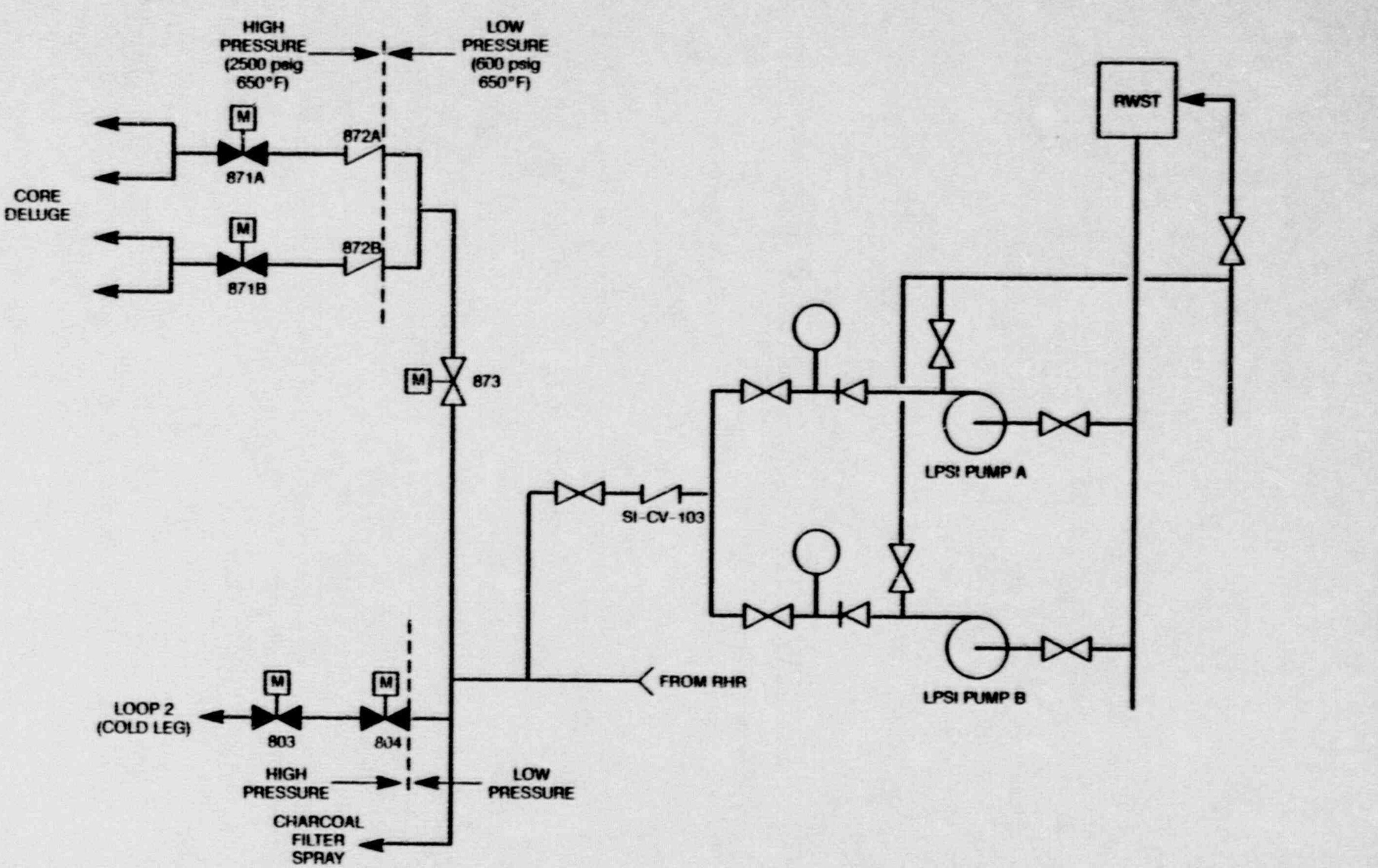

Figure 4.5 – LOW PRESSURE SAFETY INJECTION AND CORE DELUGE

CLASS	PRESSURE RATING (PSI @ 650°F)
2501R	2500
601R	600

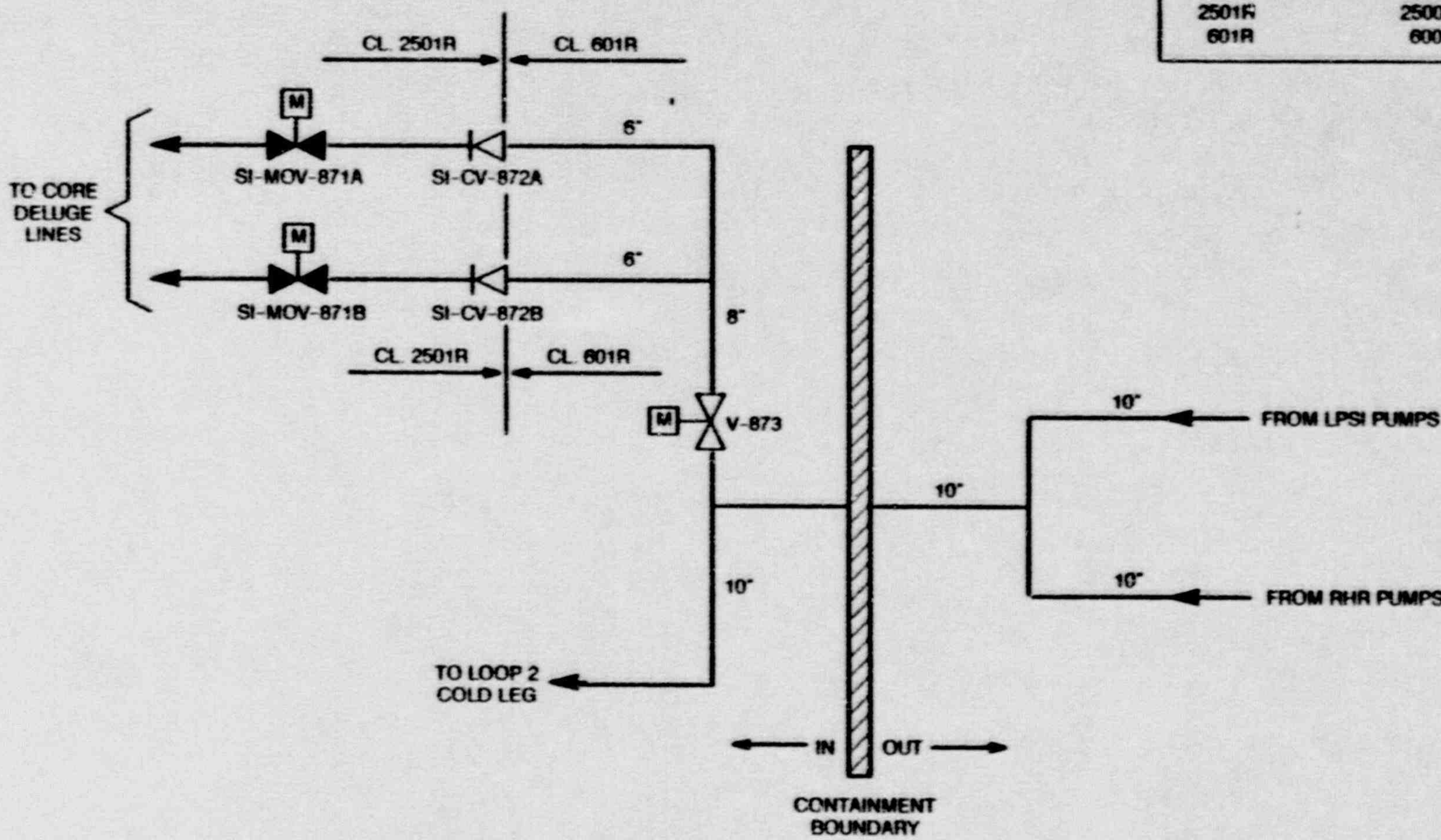

Figure 4.6 – HIGH/LOW PRESSURE INTERFACE – LPSI/CD SYSTEM

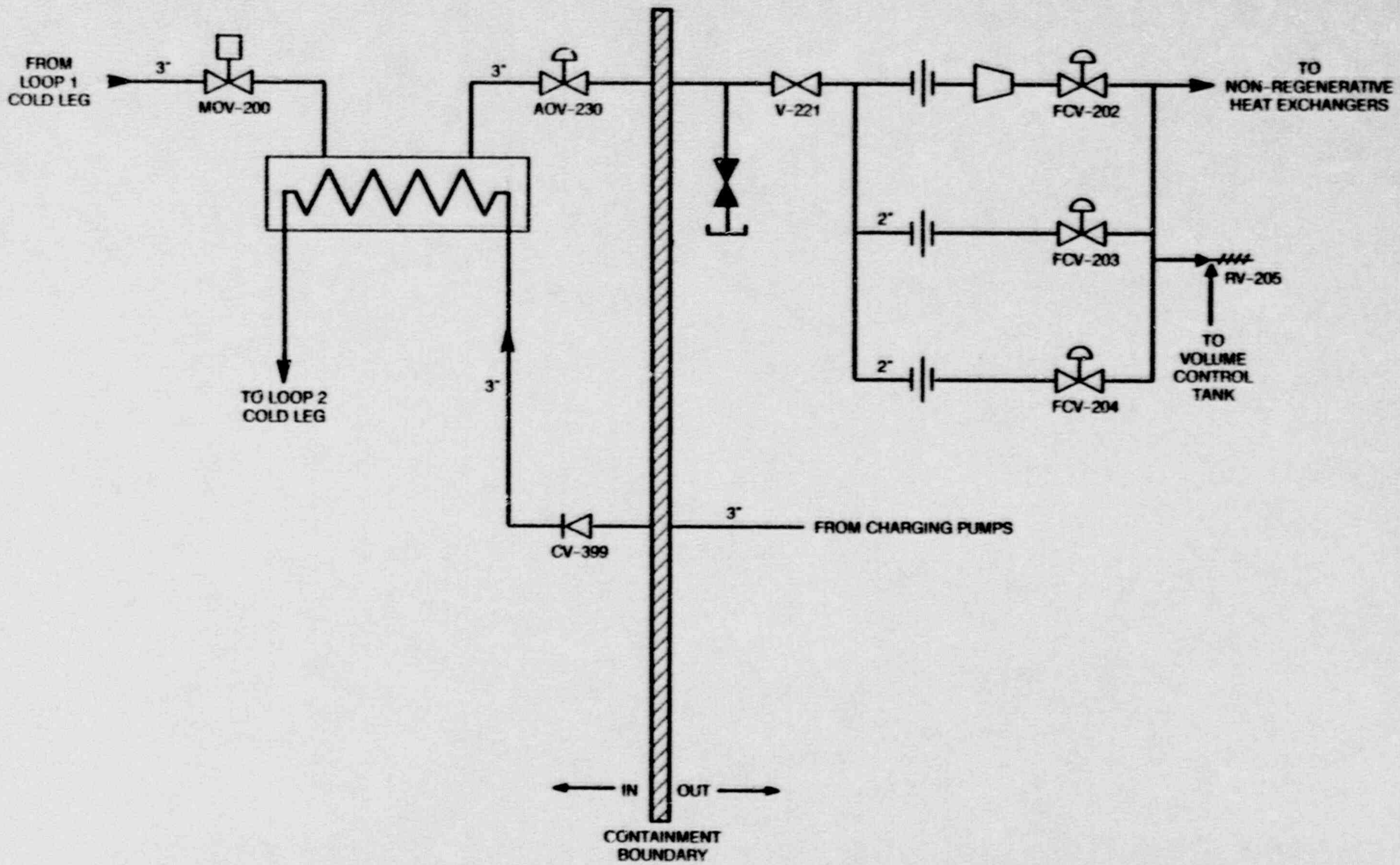

Figure 4.7 – SIMPLIFIED SCHEMATIC OF POTENTIAL PATHS FOR LOCA OUTSIDE CONTAINMENT IN THE CHEMICAL AND VOLUME CONTROL SYSTEM

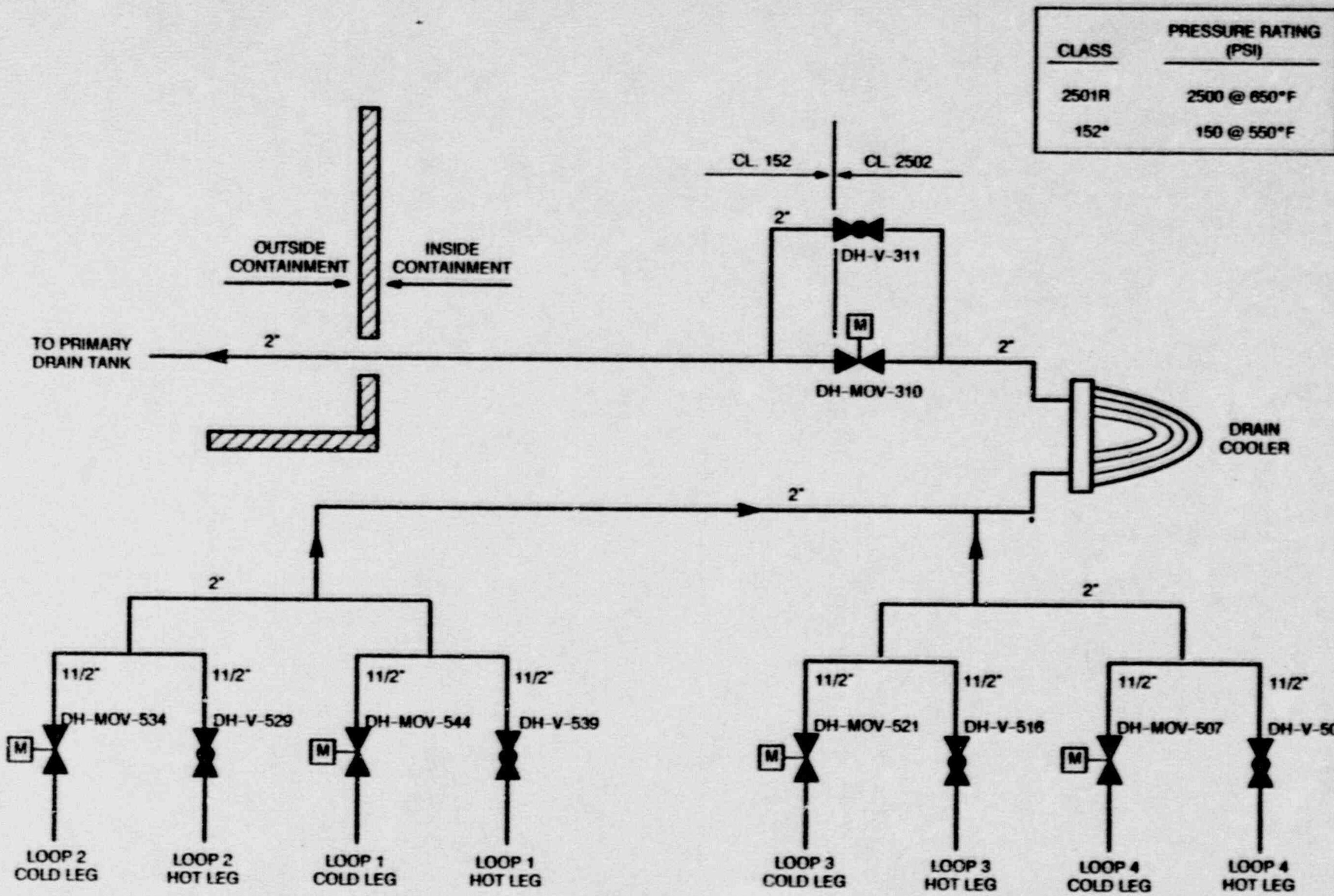

Figure 4.8 – HIGH/LOW PRESSURE INTERFACE – ALTERNATE LETDOWN LINE

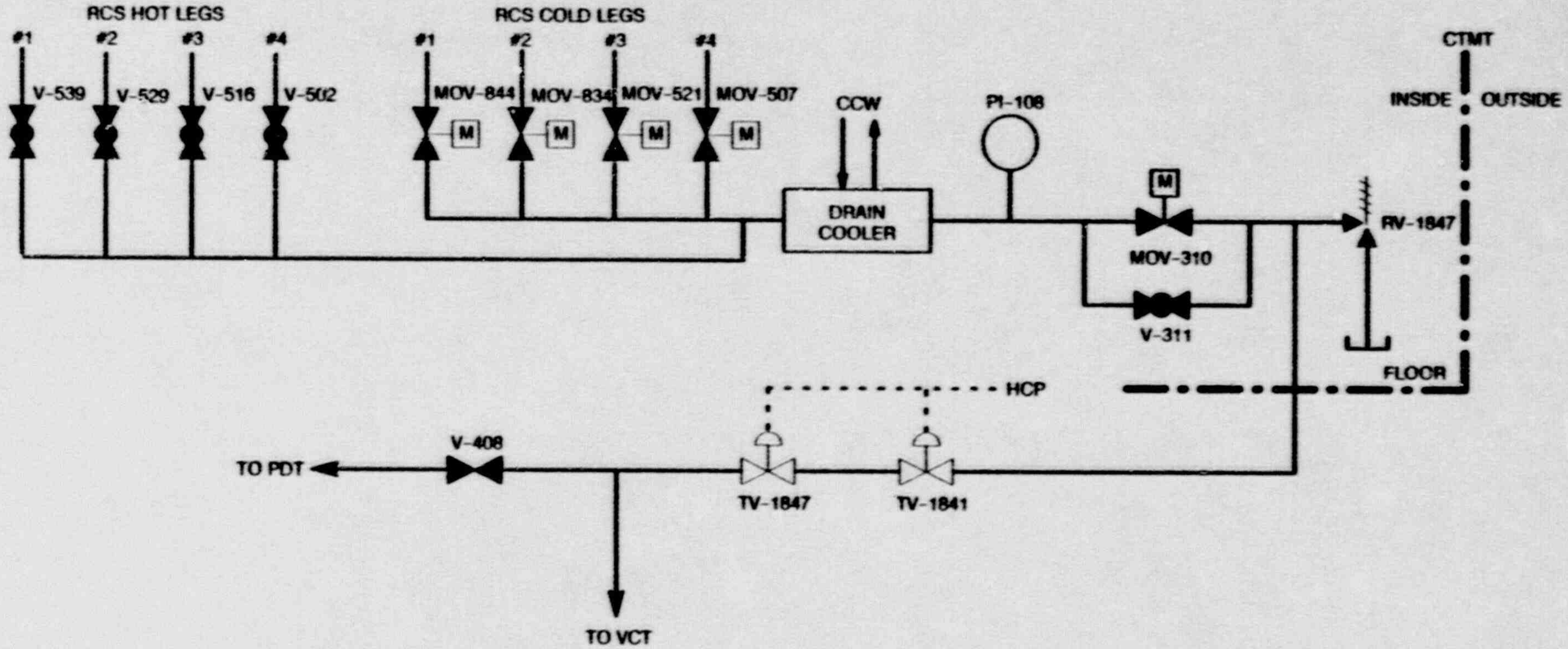

Figure 4.9 – ALTERNATE LETDOWN FLOWPATH

5.0 VALVE AND SYSTEM INTEGRITY

5.1 Maintenance Program

During the audit, the team gathered information pertinent to the mechanical maintenance history of the PIVs. This information has been tabulated on the following pages as Table 5-1. The Table contains valve number and function, a column indicating whether the maintenance received was corrective or preventive, the date the work order for the maintenance was completed, unit status at the time the work was performed, a brief description of the maintenance performed, and the cause for the maintenance. The information gathered represents the total amount of maintenance history stored on the PMMS database for the pressure isolation valves. In some cases, the information dates back as far as 1984. Of special interest are check valve SI-CV-862B in the HPI system, and power operated valve SI-MOV-861B which is in-series with this check valve. These are the PIVs discussed in Section 5.2 of this report which both demonstrated unacceptable leakage on April 6, 1986.

These two valves constitute a pressure interface boundary between the HPI pump discharge piping and high pressure piping to the RCS loop 2 cold leg. The maintenance history showed that the check valve failed numerous leak tests and received extensive internal rework before satisfying its leakage criteria on April 20, 1986. This same valve failed to satisfy its leak test acceptance criteria again on July 30, 1987, and again received internal rework (see Table 5.1 for details) in September 1987. Within the September 1987 maintenance work order was a recommendation to replace the valve because of the extent and repetitive nature of the internal rework the valve had required, the condition of the valve seat, and the fact that the valve had failed numerous leak tests. The team determined during the audit that the maintenance department did not have plans to replace this valve in the upcoming outage. This has led to some negative conclusions regarding the corrective maintenance program from an ISLOCA standpoint. The fact that the valve had a history of maintenance and testing problems, was recommended to be replaced, yet was not replaced or scheduled to be replaced is indicative of a lack of formalized performance trending of equipment which is extended to represent an overall weakness in the corrective maintenance program. Also, given the importance of the pressure isolation function of this valve, in conjunction with SI-MOV-86B which also failed leakage criteria in April 1986, this is indicative of an unawareness of ISLOCA.

As stated in Section 3.2.1, preventive maintenance (of motor operated PIVs) and post-maintenance functional testing were found to be adequate.

5.2 Inservice Testing (IST) Program

The Haddam Neck Valve IST Program, Revision 1, dated 10/3/88 contains various tables in its Appendices; each of which specifies different categories of valves. Table IWV-4, "Pressure Isolation Valves", lists the valves the Licensee considered Pressure Isolation Valves (PIVs) within its program. A total of twenty-two (22) valves were identified as PIVs on this table. The identification numbers of these valves as well as other pertinent information are shown in Table 5.3.

The NRC review of the IST program consisted of two phases. The first involved performing a review to determine if all valves which should be considered PIVs are indeed identified within the Valve IST Program as such, and the second involved determining whether the types of testing and frequencies the PIVs undergo conform to the requirements of the Code of record, which in this case is ASME Section XI, Subsection IWV, 1983 Edition.

5.2.1 Valves Considered PIVs Within the IST Program

To determine if all applicable valves were included in Table IWV-4 of the Valve IST Program, the NRC inspector reviewed the Piping and Instrumentation Diagrams (P&ID) for the Reactor Coolant System (RCS), Safety Injection System (SI), Residual Heat Removal System (RHR), and Chemical Volume Control System (CVCS) to identify the location of all potential high/low pressure interfaces between the primary system and these other systems. The P&IDs reviewed included the following:

Table 5.2

Piping and Instrumentation Diagram
(P&ID)

Drawing Number	Sheet	Rev.	Title
16103-26007	1 of 3	9	P&ID Reactor Coolant System Loops 1 & 2
16103-26007	2 of 3	8	P&ID Reactor Coolant System Loops 3 & 4
16103-26007	3 of 3	6	P&ID Reactor Coolant System Pressurizer
16103-26010	1 of 1	29	P&ID Safety Injection System
16103-26078	1 of 1	13	P&ID Residual Heat Removal System
16103-26018	1 of 8	11	P&ID Chemical & Volume Control - Letdown to Volume Control Tank
16103-26018	2 of 8	11	P&ID Chemical & Volume Control - Purification
16103-26018	3 of 8	14	P&ID Chemical & Volume Control - Boric Acid Mix System
16103-26018	4 of 8	8	P&ID Chemical & Volume Control - Charging & Metering Pumps
16103-26018	5 of 8	3	P&ID Chemical & Volume Control - Return Line to Reactor Coolant Pump Seals
16103-26018	6 of 8	4	P&ID Chemical & Volume Control - Return & Drain Lines for Reactor Coolant Loops
16103-26018	7 of 8	2	Operations Flow Diagram - Chemical & Volume Control System
16103-26018	8 of 8	3	Operations Flow Diagram - Chemical & Volume Control System

Upon identifying the location of all high/low pressure interfaces, the NRC team then reviewed the Valve IST Program to determine whether two isolation valves in series exist at each of these locations and whether these isolation valves were considered PIVs within the Licensee's program.

The NRC team concurred that all twenty-two (22) valves identified as PIVs by the licensee in Table IWV-4 should be classified as PIVs. The numbers of these valves as well as other pertinent information are listed in Table 5.3. Each of these valves can be considered to be one of the two valves acting in series that constitute the pressure isolation boundary between the primary system and other systems with a lower pressure rating.

There are, however, a series of valves which, though included in the Valve IST Program, are not considered PIVs by the program. These valves are considered the inboard pressure isolation boundary between the RCS and the lower pressure rated Drain Lines. The numbers of these valves as well as other pertinent information are listed in Table 5.3. Discussion of these valves is contained in paragraphs 5.2.2 and 5.2.3.

5.2.2 MOVs Not Considered PIVs Within the IST Program

Valve Nos. DH-MOV-544, 534, 521, and 507 are gate type motor operated valves located on the Drain Lines which come off each of the four (4) Cold Legs of the RCS. Under the Haddam Neck

Valve IST Program, these four (4) MOVs are Category B valves. Per ASME Section XI, Subsection IWV, Article IWV-2200 (a), Category B valves are those for which seat leakage in the closed position is inconsequential for fulfillment of their function. Because they are Category B valves, the only testing they are subjected to are full-stroke exercising, stroke time measurement, and observation of actual valve position.

These valves are considered the inboard pressure isolation boundary between the RCS and lower pressure piping, with either valve DH-MOV-310 or DH-V-311 being considered the outboard isolation boundary. The concern here is that even though these valves normally undergo full stroke exercising, stroke time testing, and verification of valve position every cold shutdown or refueling, they are not leak tested. The best indicator of valve integrity with respect to pressure isolation would be satisfactory leak tests results. Additionally, Category B valves are defined as those in the closed position where seat leakage is inconsequential. If a valve is to be considered the pressure isolation boundary, seat leakage should be a prime concern.

One additional concern involves the opening of these MOVs at a certain frequency to take chemistry samples. The process of opening and closing these valves frequently increases the possibility that they will not reseat properly. If the valve does not reseat properly, the effectiveness of the valve to perform as a pressure isolation boundary could be greatly hindered.

5.2.3 Other Valves Not Considered PIVs Within the IST Program

Valve Nos. DH-V-539, 529, 516, and 502 are hand operated globe valves which are categorized as passive valves under the Haddam Neck Valve IST Program (see Table 5.4). Passive valves, as defined by Subsection IWV, Article IWV-2100 are valves which are not required to change position to accomplish a specific function. Furthermore, these passive valves are not required to undergo any specific testing.

These valves are considered the inboard pressure isolation boundary between RCS and lower pressure piping, with either valve DH-MOV-310 or DH-V-311 being considered the outboard isolation boundary. They are normally opened during refueling outages to drain the steam generators. The concern here is that because these valves are categorized as passive and undergo no testing, the integrity of these valves with respect to pressure isolation is indeterminate. These valves have been in existence at the plant for over twenty (20) years and have been opened periodically; however, because they are categorized as passive, there is no way to determine to what extent they are performing their intended function as isolation valves.

One additional anomaly is the fact that Valve Nos. SI-V-863A, 863B, 863C, and 863D, which are on the Safety Injection Test Recirculation Lines, are categorized as Passive by the IST Program yet are included on Table IWV-4 as PIVs and do undergo leak testing.

5.2.4 Core Deluge Valve Nos. SI-MOV-871A and 871B

These two Core Deluge valves (SI-MOV-871A and 871B) are listed on Table IWV-4 of the Haddam Neck Valve IST Program as PIVs. Furthermore, since they are Categorized as Type A valves per ASME Section XI, Subsection IWV, they must undergo full stroke exercising, stroke time measurement, leak testing, and valve position indication. These valves, in conjunction with Valve Nos. SI-CV-872A and 872B, are considered, by the Licensee, to be the two pressure isolation boundaries between each of the two Core Deluge lines and the Low Pressure Injection System. Because of the configuration of valves SI-MOV-871A and SI-CV-872A together and SI-MOV-871B and SI-CV-872B together, the Licensee identified that it is physically impossible to leak test the two MOVs. The two valve bodies (871A and 872A) are welded together. To test reverse leakage against the check valves (872A and 872B), the MOVs can be opened to isolate the check valves themselves to get accurate test data. The only way to perform a leak test on the MOVs would be to get flow from the LPI system, through the check valves, and up against the closed MOV; however, due to the close proximity of these valves to the reactor head, physically there is no adequate means to measure the amount of leakage across each individual MOV.

TABLE 5.1

Pressure Isolation Valve Maintenance History

Ledgend

P - Power Operation
H - Hot Shutdown
C - Cold Shutdown
R - Refueling
PM - Preventive Maintenance
CM - Corrective Maintenance

TABLE 5.1

PRESSURE ISOLATION VALVE MAINTENANCE HISTORY

Valve Number	PM/CM	Date	Unit Status	Description	Cause
-MOV-803 (Loop 2 Outboard R isolation)	CM	03/18/86	R	Removed cabling and conduit from old valve. Reconnected cable and conduit to new valve. Set limit switches to CMP 8.5 - 25. Recorded start, run, and torque currents and bus volts.	Safety-related valve not EQ qualified.
	PM	10/03/87	R	Performed PMP 9.5 - 4. Inspected rotor with a boroscope per GSP-87-154 was sat. Tested MOV sat. Checked motor pinion grease level, OK.	PM
	Other	12/01/87	R	Removed T-drain and placed bracket under motor, reinstalled T-drain 88-170.	NRC EQ inspection.
I-MOV-804 (Loop 2 Inboard HR isolation)	CM	03/18/86	R	Disconnected cabling and coduit from the old valve. Reconnected cabling and conduit to new vavle. Set limits per CMP 8.5 - 25. Recorded voltage and current, timed the valve.	Safety-related valve not EQ qualified.
	PM	10/03/87	R	Boroscope inspection of rotor, was sat. Performed PMP 9.5-4. Tested MOV sat. Checked motor pinion grease level, OK.	PM

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
-MOV-780 (Loop 1 Inboard R isolation)	CM	03/18/86	R	Removed cabling and conduit from the old valve. Reconnected cabling and conduit to new valve. Set limits per CMP 8.5 - 25. Recorded start, run, and torque currents, bus volts.	Safety-related valve not EQ qualified.
	PM	10/03/87	R	Performed PM per PMP 9.5 - 4. Reliability engineering inspected rotors with boroscope per GSP-87-154. No problems noted. Tested sat. Checked motor pinion grease level, OK.	PM
-MOV-781 (Loop 1 Outboard R isolation)	CM	03/18/86	R	Removed cable and conduit from old valve. Reconnected cable and conduit to new valve. Set limits per CMP 8.5 - 25. Recorded start, run, torque, currents, and bus volts.	Safety-related valve not EQ qualified.
	PM	10/03/87	R	Performed PMP 9.5 - 4. Inspected rotor with boroscope per GSP-87-154. Tested MOV sat. Checked motor pinion grease level, OK.	PM
	Other	12/01/87	R	Removed T-drain and placed bucket under motor. T-drain installed on 88-169.	NRC EQ inspection

TABLE 5.1.
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
-MOV-871A (Core Deluge stem)	CM	02/27/85	R	Add jumpers, set limit switches and red lined control circuit, recorded start, run, and torque currents.	Safety-related valve not EQ qualified.
	Other	10/29/87	R	Perform MOVATS test.	MOVATS
	CM	02/28/88	R	Adjust packing.	None stated.
	PM	02/28/88	R	Performed PMP 9.4, inspect and lubricate.	PM
I-MOV-871B (Core Deluge stem)	CM	03/04/86	R	Add jumpers, set limit switches and red lined control circuit. Recorded start, run, and torque currents.	Safety-related valve not EQ qualified.
	CM	04/09/86	R	Moved grease relief fitting to high point on operator.	Grease relief valve in wrong location on operator.
	Other	10/28/87	R	Installed blank flange, bolts, and nuts. Placed items in hot shop to be deconed.	Per SPL 10.7 - 326 verify closure of valve.
	Other	10/29/87	R	Performed MOVATS testing.	MOVATS
	CM	02/28/88	R	Adjust packing.	

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
1-CV-872B (Core Deluge heck Valve)	CM	10/27/84	C	Disassembled and removed gasket debris. Reassembled with new seal ring.	Gasket debris in valve.
I-CV-862A (Loop 1 Safety njection check valve)	Other	09/18/87	R	Performed visual examination of valve and flange bolting.	Visual examination required per CY/ISI program.
	CM	10/05/87	R	Removed bonnet. Scotch brite seating surfaces. Cleaned out water soluable dam material. Installed bonnet and new gasket.	Valve failed leak test due to water soluable dam material inside valve.
I-CV-862B (Loop 2 cold leg afety injection valve) 3" Crane swing check valve)	CM	04/21/86	R	Disassembled, ground seats, and weld repaired seats. Reassembled flapper arm and blue checked sat. on 4/16/86. On 4/20/86, reissued for rework. Disassembled, inspected, weld repaired flapper arm, blue checked, sat.	Excessive seat leakage during ILRT (15 psig) and subsequent LLRT (40 psig) due to wear on flapper arm.
	CM	09/06/87	R	Disassembled valve, replaced swing arm and shaft, installed new disc, lapped disc. Seat was a little bit hammered. Ground seat to fine finish, lapped seat to disc. Blue checked and got 360 degree contact.	Valve has failed numerous penetration tests and has been reworked internally twice by Atlantic Valve. Note: Records indicate a recommendation to replace the valve because of worn seat.

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
	CM	09/18/87	R	Loosened all bolts but one. Removed cover and found a cloth rag stuck between disc and seat. This debris is water soluable dam material from upstream valve installation job. Continued filling with water from downstream. Lapped seats and blued. Got good contact. Replaced gasket and reassembled valve. Valve passed LLRT.	Valve failed numerous penetration tests after rework.
I-CV-862C (Loop 3 Safety njection valve)	Other	09/18/87	R	Performed visual examination of valve and flange bolting.	Visual examination required per CY/ISI program.
I-CV-862D (Loop 4 Safety njection check valve)	Other	09/18/87	R	Performed visual examination of valve and flange bolting.	Visual examination required per CY/ISI program.
I-MOV-861A (Loop 1 Safety njection stop valve)	CM	05/20/86	R	Disconnected cable and conduit from old MOV, reconnected conduit to new MOV, Raychem motor leads, set limit switches per CMP 8.5 - 25 and recorded start, run, torque currents, and bus voltage.	Safety-related valve not EQ qualified.

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
	Other	07/22/87	H	Perform MOVATS O/P Flow Test. Removed spring pack dust cover. Installed thrust measuring device. Recorded open and closed valve operation. Reassembled valve.	MOVATS
	Other	08/15/87	H	Connected MOVATS. Tested equipment and recorded as-found signatures. Left equipment as-found.	MOVATS
	PM	10/03/87	R	Performed PMP 9.5 - 4, inspect and lubricate.	PM
	CM	10/29/87	R	Adjusted packing. Cleaned off boric acid build up from valve packing.	None stated.
	CM	03/08/88	R	Performed MOVATS at MCC-5. Motor lead unit test (MLU).	MOVATS
I-MOV-861B (Loop 2 HPI top valve)	CM	04/21/86	R	Cleaned and tightened packing.	Dirty and loose packing.
	CM	04/16/86	R	Installed Raychem on motor leads and moved T-drains to proper location.	T-drains in wrong location, motor leads need Raychem.
	CM	04/23/86	R	Added three jumpers to the valve. Red lined the control circuit and set the limit switches.	Safety-related valve not EEQ qualified.

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
	Other	08/05/87	H	Performed MOVATS, found a spring pack gap, could not remove gap, reset torque switches.	MOVATS
	Other	09/20/87	R	Performed MOVATS in conjunction with ECCS flow test. ECCS test stopped due to flow equipment failure.	MOVATS
	CM	08/05/87	R	Disassembled and bled spring pack, cleaned and reassembled.	Spring pack compressed per MOVATS test finding.
	PM	08/06/87	R	Performed PM (inspect and lubricate) per procedure PMP 9.5-4.	Performance of PM
	Other	03/08/87	R	Performed MOVATS	MOVATS
1-MOV-861C (Loop 3 safety njection isolation)	CM	04/08/86	R	Replaced body to bonnet gasket. Cleaned off boric acid build up.	Valve had body to bonnet leak.
	CM	04/16/86	R	Installed Raychem on motor leads and moved T-drains to proper location.	T-drains in wrong location. Motor leads needed Raychem for EQ requirement on IKV cable.

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
	CM	04/24/86	R	Disconnected cables, reconnected cables and added three jumpers, set limit switches per CMP 8.5-25. Red lined control circuit.	Safety-related valve not EQ qualified.
	Other	09/23/87	H	Performed MOVATS testing. Found valve backseating, lots of grease in spring pack.	MOVATS testing
	PM	10/08/87	R	Performed PMP 9.5 - 4, inspect and lubricate.	PM
-MOV-861D (Loop 4 safety jection isolation)	CM	10/19/84	R	Replaced body to bonnet gasket, torqued bolts, took up on packing.	Body to bonnet leak.
	CM	04/16/86	R	Installed Raychem on motor leads and moved T-drains to proper location.	T-drains in wrong location. Motor leads need Raychem.
	CM	04/29/86	R	Removed cabling, wired jumpers and reconnected cabling, set limit switches per CMP 8.5 - 25 and red-lined control circuit.	Safety-related valve not EQ qualified.
	Other	08/22/87	H	Performed MOVATS on valve. Found valve backseating.	MOVATS

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
	PM	10/08/87	R	Performed PMP 9.5 - 4, inspect and lubricate.	PM
	CM	03/08/88	R	Performed MOVATS motor lead unit test.	MOVATS
H-MOV-310 (RCS drain cooler utlet valve)	PM	04/22/86	R	Disassemble, inspect, repair, and lubricate.	PM
	PM	08/14/87	R	Inspect and lubricate. Record number of limit switch rotors.	PM
	CM	10/16/87	R	Torque switch setpoints changed from 1.0 open and closed to 2.0 open and closed.	Change torque switch settings.
	CM	10/21/87	R	Changed torque switch setting to 4.5	Change torque switch setting.
	CM	01/07/88	R	Investigate and test control circuit as necessary. Verified wiring as noted on drawings.	Problem with control circuit.
H-V-539 (Loop 1 RCS drain alve)	CM	08/25/84	R	Repack gland, clean off boric acid on stem, tighten packing.	Boric acid on packing gland.
H-V-539A (Loop 1 RCS drain alve)	CM	09/08/84	C	Repack valve	Packing gland bolts backed out and packing coming out of stutting box.

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
H-MOV-544 (Loop 1 MOV drain eader isolation valve)	PM	03/19/86	R	Disassemble, inspect, repair, and lubricate.	PM
	CM	04/19/86	R	Reterminate unsatisfactory terminations.	None stated.
	Other	09/18/87	R	Perform visual examination of valve and flange bolting.	ISI requirement for visual examination.
	PM	10/01/87	R	Disassemble, inspect, repair, and lubricate.	PM
	PM	10/29/87	R	Replace breaker.	PM
)H-V-529 (Loop 2 RCS drain valves)	CM	08/25/84	C	Repack valve, clean off boric acid, tighten gland	Boric acid leaking through packing.
)H-MOV-534 (Loop 2 MOV drain valve)	PM	03/11/86	R	Disassemble, inspect, repair, and lubricate.	None stated.
	Other	09/18/87	R	Perform visual examination of valve and flange bolting.	Visual examination required per ISI program.
	PM	10/01/87	R	Disassemble, inspect, repair, and lubricate.	None stated.
)H-V-516 (Loop 3 RCS drain valve)	CM	05/06/88	R	Adjust packing as necessary.	Packing leak.
)H-MOV-521 (Loop 3 drain valve)	CM	08/31/84	R	Replaced gland eye bolts.	Gland eye bolts severly corroded.
	PM	03/19/86	R	Disassemble, inspect, repair, and lubricate.	PM

TABLE 5.1
(CONTINUED)

Valve Number :	PM/CM :	Date :	Unit Status :	Description :	Cause :
	Other	09/18/87	R	Perform visual examination of valve and flange bolting.	Visual examination required per CY ISI program.
	PM	10/01/87	R	Disassemble, inspect, repair, and lubricate.	PM
H-MOV-507 (Loop 4 drain valve)	PM	03/18/86	R	Disassemble, inspect, repair, and lubricate.	PM
	PM	10/01/87	R	Disassemble, inspect, repair, and lubricate.	PM
	CM	03/05/88	C	Adjust packing as necessary.	Packing leak.
H-V-502 (Loop 4 RCS drain valve)	CM	09/22/84	C	Tighten down gland follower.	None stated.
	CM	03/20/86	R	Adjust packing.	None stated.

TABLE 5.1
(CONTINUED)

MAINTENANCE HISTORY

Valve Number	PM/CM	Date	Unit Status	Description	Cause
D-AOV-230 (Let down header rip valve)	CM	09/19/87	R	Adjusted packing that was leaking during hydro.	New valve packing wasn't adjusted.
	CM	01/28/88	R	Replaced diaphragm.	Manufacturer suggestion to change diaphragm. Note: Work was delayed due to poor procedure.
D-MOV-200 (Let down iso. to egen Hx)	CM	04/16/86	R	Installed Raychem on motor leads and moved T-drains to proper location.	Safety-related valve not EEQ qualified. Also, T-drains were in wrong location.
	CM	05/20/86	R	Tested new overload relay with new O/L heater, replaced old O/L relay with new one.	EQ
	CM	05/20/86	R	Disconnected cables, reconnected cables, added jumpers, set limit switches, red lined the control circuit.	EQ
	CM	05/20/86	R	MOV overloads are undersized for the motor current drawn. Installed new O/L heaters per DCN 80-201-1, tested heaters per M8.5 - 126.	Wrong size installed per PA 80-201.
	CM	07/30/86	H	Adjusted pacing. Leak off line seems to have cooled down.	Packing out of allignment.

TABLE 5.1
(CONTINUED)

Valve Number	PM/CM	Date	Unit Status	Description	Cause
	PM	10/06/87	R	Performed PM per PMP 9.5 - 4. Inspected and lubriaced ME-51 limitorque.	PM
)-MOV-200	CM	01/23/88	R	Removed motor, disassembled, removed grease, reassembled, installed and tested. New limit switch gasket installed. Removed space heater from motor.	Grease in motor.
	CM	02/09/88	R	Removed foreign material from between closed torque switch contacts. (Appeared to be pipe covering insulation material.)	Valve would not cycle electrically.
	CM	03/05/88	R	Adjusted packing, tightened down body to bonnet bolts.	Packing and body to bonnet bolt leak.

The Haddam Neck Valve IST Program addresses this particular situation as follows:

"CD-MOV-871A/B and CD-CV-872A/B are welded to one another with no provision for leak testing CD-MOV-871A/B using reactor coolant pressure.

At system hydrotest at startup, CD-MOV-871A/B are open thus subjecting CD-CV-872A/B to reactor coolant pressure. Leakage is measured at this time."

Note: P&ID No. 16103-26028, "Residual Heat Removal System" identifies these valves as "SI" for Safety Injection while the Valve IST Program for the second interval identifies them as "CD" for Core Deluge. They refer to the same valves.

The situation that exists here is similar to that previously identified for Valve Nos. DH-MOV-544, 534, 521, and 507. The valves (DH-MOV-871A/B and DH-CV-872A/B) which constitute the two pressure isolation boundaries between the RCS and a lower pressure system (in this case LPI) have been identified within the Valve IST Program; however, one of the two valves in series (in this case MOVs DH-MOV-871A/B) is not subjected to leak testing to ensure valve integrity.

5.2.5 Valve No. SI-MOV-873

Presently, this valve is a hand operated gate valve located inside containment just outboard of the core deluge valves SI-MOV-871A/B and SI-CV-872A/B. During the 1989 outage, the Licensee will change the actuation of this valve to a motor operated type , controllable from the Control Room. Changing the actuation of this valve will greatly increase the effectiveness in mitigating an ISL on the core deluge line. If an ISL were to occur due to the failure of the two core deluge PIVs in series, valve SI-MOV-873 would be available to either stop the loss of inventory outside containment or to confine the loss of inventory to inside the containment.

TABLE 5.3

PRESSURE ISOLATION VALVES IDENTIFIED WITHIN THE HADDAM NECK IST PROGRAM

Valve No.	Class Category	Function	Size	Type	Actuation	Normal Position	Test Required
SI-MOV-861A	1-A	Loop 1 Safety Injection Isol.	3	GA	Motor	Closed	Q MT LT PI
SI-MOV-861B	1-A	Loop 2 Safety Injection Isol.	3	GA	Motor	Closed	Q MT LT PI
SI-MOV-861C	1-A	Loop 3 Safety Injection Isol.	3	GA	Motor	Closed	Q MT LT PI
SI-MOV-861D	1-A	Loop 4 Safety Injection Isol.	3	GA	Motor	Closed	Q MT LT PI
SI-CV-862A	1-AC	Loop 1 SI Isol. Check - P3	3	CK	-	Closed	CV LT
SI-CV-862B	1-AC	Loop 2 SI Isol. Check - P3	3	CK	-	Closed	CV LT
SI-CV-862C	1-AC	Loop 3 SI Isol. Check - P3	3	CK	-	Closed	CV LT
SI-CV-862D	1-AC	Loop 4 SI Isol. Check - P3	3	CK	-	Closed	CV LT
SI-V-863A	2-A	Loop 1 Test Recirc. to RWST	.75	GL	Hand	Locked Closed	LT
SI-V-863B	2-A	Loop 2 Test Recirc. to RWST	.75	GL	Hand	Locked Closed	LT
SI-V-863C	2-A	Loop 3 Test Recirc. to RWST	.75	GL	Hand	Locked Closed	LT
SI-V-863D	2-A	Loop 4 Test Recirc. to RWST	.75	GL	Hand	Locked Closed	LT
SI-MOV-871A	1-A	Core Deluge to RV Head Motor	6	GA	Motor	Closed	Q MT LT PI
SI-MOV-871B	1-A	Core Deluge to RV Head Motor	6	GA	Motor	Closed	Q MT LT PI

TABLE 5.3 (continued)

Valve No.	Class Category	Function	Size	Type	Actuation	Normal Position	Test Required
SI-CV-872A	1-AC	Core Deluge to RV Head Check	6	CK	-	Closed	CV LT
SI-CV-872B	1-AC	Core Deluge to RV Head Check	6	CK	-	Closed	CV LT
RH-MOV-780	1-A	Inboard Stop RCS Loop #1	10	GA	Motor	Closed	Q MT LT PI
RH-MOV-781	1-A	Outboard Stop RCS Loop #1	10	GA	Motor	Closed	Q MT LT PI
RH-MOV-803	1-A	Outboard Stop RCS Loop #2	10	GA	Motor	Closed	Q MT LT PI
RH-MOV-804	1-A	Inboard Stop RCS Loop #2	10	GA	Motor	Closed	Q MT LT PI
DH-MOV-310	1-A	Drain Header Remote Throttle	2	GA	Motor	Closed	Q MT LT PI
DH-V-311	1-A	Drain Header	2	GA	Hand	Closed	LT

Where:

- GA - gate valve
- CK - check valve
- GL - globe valve
- Q - Exercise valve (full stroke) to verify satisfactory operation per ASME Section XI, IWV-3411 and IWV-3521.
- LT - Valves are leak tested per Section XI, Article IWV-3420.
- MT - Stroke time measurements are taken per Section XI, Article IWV-3410 for power operated valves.
- CV - Exercise check valves to the position required to fulfill their function per Section XI, Article IWV-3521 except as noted in IWV-3522.
- PI - Visually observe, every two (2) years or less, actual valve position to confirm that remote valve position indications accurately reflect valve operation, IWV-3300. Examples of how this can be done are: verifying local position indicator, or flow, or pressure change, or stem traveling in the correct direction.

TABLE 5.3 (Continued)

Notes For Table 5.3

1. Column "Class/Category" lists the ASME Section III code class of the valve as designated by the Licensee and the ASME Section XI, Article IWV-2200 category of the valve.

2. Column "Test Required" lists the testing requirements of Section XI, Subsection IWV; which is not neccesarily the test performed by the Licensee.

3. Valves SI-MOV-861A, 861B, 861C, SI-MOV-871A, 871B, RH-MOV-780, 781, 803, 804, and DH-MOV-310 are full stroke exercised only during cold shutdown.

4. Valves SI-CV-862A, 862B, 862C, and 862D are partially stroked every quarter and full stroked every refueling per Relief Request No. V-2.

5. Valves SI-CV-862A, 862B, 862C, and 862D are leak tested every cold shutdown.

6. Valves SI-MOV-871A and 871B are not leak tested per Relief Request No. V-6. Each refueling, the valves shall be verified fully closed by utilizing MOVATs.

7. Valves SI-V-863A, 863B, 863C, 863D, and DH-V-311 are classified as Passive valves by the IST program.

8. Valve DH-V-311 is listed as a gate valve in the IST Program and, however, it is depicted as a globe valve on drawing No. 16103-26007, sheet 3 of 3, "P&ID RCS Pressurizer".

TABLE 5.4

VALVES NOT LISTED AS PIVs BY THE HADDAM NECK IST PROGRAM

Valve No.	Class Category	Function	Size	Type	Actuation	Normal Position	Test Required
DH-MOV-544	1-B	Loop 1 Drain/ Alt. Letdown	1.5	GA	Motor	Closed	Q MT PI
DH-MOV-534	1-B	Loop 2 Drain/ Alt. Letdown	1.5	GA	Motor	Closed	Q MT PI
DH-MOV-521	1-B	Loop 3 Drain/ Alt. Letdown	1.5	GA	Motor	Closed	Q MT PI
DH-MOV-507	1-B	Loop 4 Drain/ Alt. Letdown	1.5	GA	Motor	Closed	Q MT PI
DH-V-539	1-B	Loop 1 Drain/ Alt. Letdown	1.5	GL	Hand	Closed	ET
DH-V-529	1-B	Loop 2 Drain/ Alt. Letdown	1.5	GL	Hand	Closed	ET
DH-V-516	1-B	Loop 3 Drain/ Alt. Letdown	1.5	GL	Hand	Closed	ET
DH-V-502	1-B	Loop 4 Drain/ Alt. Letdown	1.5	GL	Hand	Closed	ET

Where:

- GA - gate valve
- GL - globe valve
- Q - Exercise valve (full stroke) to verify satisfactory operation per ASME Section XI, IWV-3411 and IWV-3521.
- MT - Stroke time measurements are taken per Section XI, Article IWV-3410 for power operated valves.
- PI - Visually observe, every two (2) years or less, actual valve position to confirm that remote valve position indications accurately reflect valve operation, IWV-3300. Examples of how this can be done are: verifying local position indicator, or flow, or pressure change, or stem traveling in the correct direction.
- ET - Verify and record valve position before operations are performed and after operations are completed, and verify that valve is locked or sealed.

5.3 Surveillance Program

5.3.1 Surveillance Program for MOVs

This section details the audit performed by the NRC for the Haddam Neck Surveillance Program for motor operated valves. The details of the NRC audit for the check valves surveillance program is in section 5.3.2. Each subparagraph under section 5.4.1 will describe the audit of individual groups of valves identified as PIVs by the Licensee.

5.3.1.1 Valve Nos. SI-MOV-861A, 861B, 861C, and 861D

These four (4) valves are identical 3" motor operated gate valves and they function as the Safety Injection Isolation Valves for each of the four RCS loops. These valves serve as the pressure interface between the RCS and the HPI system and they open upon receipt of an SI signal to admit water from the high pressure safety injection pumps to the RCS cold legs. The procedure review was limited to pressure isolation test (leak testing); not full stroke exercising or stroke time measurements. Assessment of the surveillance test results, however, included stroke time measurement data as well as the pressure isolation data.

Procedure No. SUR 5.7-128, "Loop Safety Injection Stop Valves SI-MOV-861A, B, C and D Pressure Isolation Test", revision 1 Major, effective date of June 6, 1989, was reviewed by the NRC inspector. The procedure deals with the periodic measurement of any leakage through loop safety injection stop valves SI-MOV-861A, B, C and D. The review included, but was not limited to the following aspects:

a) Is the proper lineup of valves prior to, during, and upon completion of the test specified to perform an accurate and safe test for each of the four valves?

b) Is the testing frequency per Section XI, Subsection IWV?

c) Utilization of appropriate and proper test equipment i.e. calibrated equipment where necessary.

d) Are correct test parameters specified? i.e. utilization of appropriate test pressures

e) Is the appropriate plant Mode of operation specified to perform this test?

f) Is the manner of data collection clearly defined as to how and when it should occur during the execution of the test?

g) Are there appropriate signoffs and approvals?

h) Is the recorded data properly extrapolated into useful units? i.e., conversion of CC/MIN @ 350 psig to GAL/MIN @ 2000 psig

The NRC team determined that the above mentioned criteria are correctly contained making for a clear, concise and technically adequate procedure.

The surveillance test results for Procedure No. SUR 5.7-128 from the last 3 1/2 years were reviewed for each of the four valves. In all instances, the test frequency met the criteria specified in the procedure. Test results were below the specified acceptable limits with the exception of one of the four valves. For valve SI-MOV-861B during the time frame April 7, 1986 to April 20, 1986, six (6) tests were run, with four out of the six failing to meet the acceptance criteria for leakage. The final test, before returning to power, did meet the specified acceptance criteria. The leak test results for valve SI-MOV-861B are as follows:

Test Date	As Found Leak Rate (GPM)	Acceptance Criteria
4/07/86	1.21	1 GPM @ 2000 psig
4/14/86	1.20	"
4/16/86	0.95	"
4/18/86	1.18	"
4/19/86	1.06	"
4/20/86	0.08	"

Further discussions on the sequence of events relative to this valve failing to meet the specified acceptance criteria are discussed in section 7.1 of this report.

It was also verified that for the past 3 1/2 years, stroke time testing had been performed during the frequencies specified by ASME Section XI, Article IWV-3410. Furthermore, the test results indicate that all four valves had met the specified acceptance criteria for stroke time testing.

5.3.1.2 Valve Nos. SI-V-863A, 863B, 863C, and 863D

These four (4) valves are identical 3/4" hand operated globe valves on the HPSI recirculation lines. These lines penetrate the cold leg injection lines between the check valve and the SI stop valves. These four valves, which are normally locked closed, isolate the recirculation line to ensure all flow is directed into each of the RCS cold legs. The only testing performed on these valves is the pressure isolation test (leak testing).

Procedure No. SUR 5.7-66, "Safety Injection Recirculation, P-24", revision 5 Major, effective date of February 17, 1989, was reviewed by the NRC inspector. The procedure deals with the periodic measurement of any leakage through safety injection recirculation valves SI-V-863A, B, C and D. The scope of this procedure review was similar to that described in the previous section for Procedure No. SUR 5.7 128.

Based upon this review, the NRC inspector determined the procedure is clear, concise and technically adequate.

The surveillance test results for Procedure No. SUR 5.7-66 from the last 3 1/2 years were reviewed for each of the four valves. In all instances, the test frequency met the criteria specified in the procedure. Test results were below the specified acceptable limits in all instances reviewed.

5.3.1.3 Valve Nos. SI-MOV-871A and 871B

These two (2) valves are identical 6" motor operated gate valves on the reactor head core deluge lines. These two valves are normally closed and will open upon actuation to allow water from the RHR pumps into the core deluge lines.

Because of the location and configuration of these valves, it is physically impossible to perform a pressure isolation test i.e. leak test. Each valve is welded to its corresponding check valve (either valve SI-CV-872A or 872B). If flow is sent to these valves from the RHR pump side, there is no way to measure the leakage of the valve because it is so close to the reactor head. If flow is sent from the reactor head side of the valve, the flow would also be acting against the check valve which is welded to this valve; therefore, an accurate determination of leakage across the valve could not be made unless the check valve was held open. This is impractical to do. Relief against leak testing was obtained by the licensee and is depicted as Relief Request V-6 in the Licensee's Valve IST Program.

It was verified, however, that for the past 3 1/2 years, stroke time testing had been performed during the frequencies specified by ASME Section XI, Article IWV-3410. Furthermore, the test results indicate that all four valves had met the specified acceptance criteria for stroke time testing.

5.3.1.4 Valve Nos. RH-MOV-780, 781, 803 and 804

These four (4) valves are identical 10" motor operated gate valves. RH-MOV-780 and 804 are the inboard isolation valves that isolate the high pressure RCS loops from the low pressure RHR system. RH-MOV-781 and 803 are the outboard isolation valves. Valves RH-MOV-780 and 781 isolate the loop 1 hot leg from the RHR pump suction. Valves RH-MOV-803 and 804 isolate the loop 2 cold leg from the RHR heat exchanger discharge. The procedure review was limited to pressure isolation test (leak testing); not full stroke exercising or stroke time measurements. Assessment of the surveillance test results, however, included stroke time measurement data as well as the pressure isolation data.

Procedure No. SUR 5.7-5.1, "Hydrostatic Test", revision 14, effective date of October 27, 1987, was reviewed by the team. The procedure deals with testing of the RCS to verify its leak tightness after the system has been opened for refueling or maintenance. Additionally, the procedure verifies the leak testing of PIVs RH-MOV-780, 781, 803 and 804. The portions of the procedure which were reviewed dealt only with the leak testing of the four isolation valves. The scope of this procedure review was similar to that described in the previo section 5.4.1.1 for Procedure No. SUR 5.7 128.

It was determined that the procedure is well written in that it was concise and clear. It was also technically adequate to determine the amount of leakage through each of the four above mentioned RHR isolation valves.

The surveillance test results for Procedure No. SUR 5.1-1 from approximately the last 5 years were reviewed for each of the four valves. In all instances, the test frequency met the criteria specified in the procedure. Test results were below the specified acceptable limits in all instances. It was also verified that for the past 3 1/2 years, stroke time testing had been performed during the frequencies specified by ASME Section XI, Article IWV-3410. Furthermore, the test results indicate that all four valves had met the specified acceptance criteria for stroke time testing.

5.3.1.5 Valve Nos. DH-MOV-310 and DH-V-311

Valve DH-MOV-310 is a 2" motor operated gate valve. Its function is to isolate and regulate the flow of the RCS drain header. Valve DH-V-311 is a 2" hand operated globe valve. Its function is to isolate the RCS drain header. The procedure review was limited to pressure isolation test (leak testing); not full stroke exercising or stroke time measurements. Assessment of the surveillance test results, however, included stroke time measurement data, if applicable, as well as the pressure isolation data.

Procedure No. SUR 5.7-46, "Loop Drains Header, P-41", revision 7 major, effective date of June 28, 1988, was reviewed by the NRC inspector. The procedure deals with the periodic testing of leakage for valves DH-MOV-311 and DH-V-310. The scope of this procedure review was similar to that described in the previous section 5.4.1.1 for Procedure No. SUR 5.7 128.

It was determined that the procedure is well written in that it was concise and clear. It was also technically adequate to determine the amount of leakage through each of the two above mentioned drain header isolation valves.

The surveillance test results for Procedure No. SUR 5.7-46 from approximately the last 3 1/2 years were reviewed for each of the valves.

It was also verified that for the past 3 1/2 years, stroke time testing for valve DH-MOV-310 had been performed during the frequencies specified by ASME Section XI, Article IWV-3410. Furthermore, the test results indicate that the valve had met the specified acceptance criteria for stroke time testing.

5.3.2 Surveillances of HPI and Core Deluge Check Valves

The audit team reviewed the surveillances of the licensee's check valves in the HPI, LPI, and RHR systems which were classified as Pressure Isolation Valves (PIVs). There PIVs provided one of the barrier between high pressure reactor coolant system piping and the lower pressure Emergency Core Cooling System (ECCS) piping. The review purpose was to determine the extent and adequacy of the surveillances currently being performed on the check valves:

SUR 5.7-135, Original, "HPSI and LPSI Discharge Check Valve Operability Test"
SUR 5.7-111, Rev.02, "Leak Testing of Core Deluge Check Valves"
SUR 5.1-4, Rev. 22, "Core Cooling Systems Hot Operational Test"
SUR 5.7-65,Rev. 00, "Loop Safety Injection P-3 Isolation Valves SI-CV-862 A, B, C, and D Local Leak Rate And Pressure Isolation Testing"

These procedures were, in general, thorough, comprehensive and well written. Not only were the check valves testeo to verify that they will pass the full required flow, but also, they were tested to ensure that back leakage was not excessive. Licensee's back-leakage acceptance criteria for these check valves was 1 gpm. There were no check valves classified as Pressure Isolation Valve in the RHR system.

Although the procedures were generally comprehensive, surveillance procedure SUR 5.7-111, "Leak Testing of Core Deluge Check Valve" required a procedural clarification in that it did not specify RHR pump operation when measuring leakage across SI-V-873. SI-V-873 is a core deluge header combined discharge valve which is presently a manual isolation valve that is scheduled to be converted into a motor-operated valve during the upcoming outage. Although SUR 5.7-111 is performed to measure leakage across SI-CV-872A & B (see Figure 4.5), leakage across SI-V-873 is also measured to ensure accurate leakage test data for the core deluge check valves. For leakage tests of SI-V-873, leak measurements are taken at the drain valves SI-V-873A & B, SI-V-8731 in parallel with the core deluge check valves. Consequently, it is important that any leakage across SI-V-873 be accurately determined to prevent excessive leak rates from being assigned to the core deluge check valves in the event SI-V-873 is leaking. Although the surveillance procedure imply the requirement for RHR pump operation, the procedure does not specify the operational requirement of the pump, and the surveillance can be performed without implementing this important prerequisite.

Additionally, the licensee was not performing internal visual examination of any of the check valves in the HPI and core deluge headers for wear and detection of internal check valve problems. Although these check valves are not routinely in service because of the systems in which they are located, they have been in service for over 20 years and some wear on the valve internals should be expected. In fact, the team found that because of the hinge mechanism failure in check valve, SI-CV-862B, located in one of the four HPI injection lines, licensee experienced a 50.9 gpm leak. This type of check valve problem could have been detected and prevented through internal examination of check valves. Furthermore, the importance of proper check valve operation and integrity is even more important in the core deluge system, where check valves, SI-CV-862A & B provide the only testable pressure isolation valve because its downstream PIVs, SI-MOV-871A & B, cannot be leak tested due to its physical proximity to the Reactor Vessel head and these check valves. Consequently, the licensee does not know whether SI-MOV-871A or SI-MOV-871B is presently in a condition to act as a pressure isolation barrier between high pressure reactor coolant system and a relatively lower high pressure injection header.

5.4 I&C Surveillances and Program

5.4.1 Reactor Coolant Pressure Channel Calibration

The audit team reviewed Instrumentation and Controls (I&C) procedure SUR 5.2-52, "Reactor Coolant Pressure Channel Calibration", used to calibrate the pressure instrument and its associated instrumentation. The instrumentation provides the pressure interlock that prevents the inadvertent opening of RH-MOV-780, 781, 803, and 804 when reactor coolant pressure is above 400 pounds.

The procedure was well written, comprehensive and no discrepancies were found. The values (RH-MOV-780, 781, 803, and 804) provide the cooldown lineup to cool RCS from 300°F to cold shutdown by providing a suction path from loop 1 hot leg via RH-mov-780, 781 and discharge path to loop 2 cold leg via RH-MOV-803 and 804. The purpose of the procedure was to calibrate the channel output of pressure transmitters P-403 and 404 such that it responds with acceptable range and accuracy. Both P-403 and 404 "As Found" values are taken and pressure instruments are calibrated if the "As Founds" do not fall within the acceptable range. Additionally, other associated instruments such as various pressure monitors, recorders, controllers, indicators are calibrated in this procedure. SUR 5.2-52 is performed at least once every 18 months and it was last performed on October 20, 1987.

5.4.2 MOV Torque Switch Settings

The audit team reviewed the licensee's control of MOV torque switches for the PIVs in the RHR, HPI and the Core Deluge Systems. The licensee appeared to have adequately implemented controls on their pressure isolation valve MOVs. Torque switch bypasses, torque switches and position limit switches were set correctly to allow proper valve operation with maximum differential pressures expected on these valves during both normal and abnormal events within the design basis. The surveillance status of the PIVs are summarized in Table 5.5.

There are currently ten motor-operated valves classified as PIVs by the licensee in the RHR, HPI and Core Deluge systems. Four of these ten MOVs in the HPI discharge line have been analyzed to determine their maximum differential pressure expected during both opening and closing the valve for both normal and abnormal events within the existing design basis. Torque switches for two MOVs in the core deluge discharge line and four RHR isolation valves have not been analyzed to determine their expected torque value associated with the maximum differential pressure expected during both opening and closing for both normal and abnormal events. This will ensure proper valve operation. Presently all six of these torque switch settings have been set to vendor recommended values. These MOVs in the RHR and the Core Deluge systems normally do not expect to see high differential pressure during normal use and were not required to be evaluated in accordance with IE Bulletin 85-03. However, because of the importance of the valves as PIVs, the audit team expressed some concern as to whether the presently set torque switch settings would be adequate for valve operation during potential postulated accident conditions.

Adjustment of Limitorque limit and torque switches are performed using licensee's maintenance procedure PMP 9.5-215.1, "Limitorque Operator Removal, Installation and Adjustment." The full open limit switch is set by fully opening the valve manually and then manually closing the valve slightly (1/4 turn to a few turns beyond engagement depending on valve size) to allow for coasting and to prevent backseating. Also, the full open limit switch is used to de-energize the valve position green light (close indication). Full close limit switch is set in a similar fashion. The valve is manually closed and then opened slightly until valve stem just starts to move, or the valve is just off the seat. Additionally, normally full closed limit switch is set to de-energize the valve position red light (open light). The torque bypass switches are set by first calculating the number of handwheel turns to obtain full stroke and then ensuring that the open and close torque switch bypasses fall within the acceptable range of valve stroke as follows:

Open Bypass Maximum	25 percent of Full Stroke
Open Bypass Minimum	15 percent of Full Stroke
Close Bypass Maximum	10 percent of Full Stroke
Close Bypass Minimum	5 percent of Full Stroke

Torque switch adjustment is made by referring to the the master setpoint list to obtain the proper setting and any adjustment made accordingly. Additionally, changes to torque switches cannot be made without an approved setpoint change request and an independent verification is required to ensure that it is set properly.

Table 5.5 PIV Surveillances

Valve #s:	Adequately Leak Tested	Adequately Stroke Tested	Torque Switch Properly Set/Controlled
RH-MOV-780	Y	Y	Ind./NEC
RH-MOV-781	Y	Y	Ind./NEC
RH-MOV-803	Y	Y	Ind./NEC
RH-MOV-804	Y	Y	Ind./NEC
SI-MOV-861A	Y	Y	Ind./NEC
SI-MOV-861B	Y	Y	Ind./NEC
SI-MOV-861C*	Y	Y	Y
SI-MOV-861D	Y	Y	Y
SI-MOV-871A	(1)	Y	Y
SI-MOV-871B	(1)	Y	Y
DH-MOV-507	N(2)	Y	Ind./NEC
DH-MOV-521	N(2)	Y	Ind./NEC
DH-MOV-534	N(2)	Y	Ind./NEC
DH-MOV-544	N(2)	Y	Ind./NEC
DH-V-502	N(2)	N/A	N/A
DH-V-516	N(2)	N/A	N/A
DH-V-529	N(2)	N/A	N/A
DH-V-539	N(2)	N/A	N/A
DH-V-311	Y	N/A	N/A
DH-MOV-310	Y	Y	Ind./NEC

Valve #s:	Full Flow	Reverse leakage	Internal Exam
SI-CV-862A	Y	Y	N
SI-CV-862B(3)	Y	Y	N
SI-CV-862C	Y	Y	N
SI-CV-862D	Y	Y	N
SI-CV-872A	Y	Y	N
SI-CV-872B	Y	Y	N

NOTES:

(1) These PIVs cannot be leak tested due to their close proximity to their upstream check valves.

(2) Presently the licensee does not consider these valves to be PIVs and therefore no leakage test is performed.

(3) SI-CV-862B has recently experienced numerous problems . This check valve was found to have 50 gpm backleakage during a routine surveillance on 04/07/86 and was reworked five times between 04/07/86 and 04/17/86 in order to bring its backleakage specification below its acceptance criteria of 1 gpm. The initial failure appeared to have been check valve hinge pin wear causing misalignment of the check valve disc and its seat. Additionally, this valve failed its leakage test again on 07/30/87 and subsequently reworked on 09/06 and 09/18/87. Licensee has not had problems with their other check valves that are considered PIVs.

Ind./NEC: Indeterminate/No Engineering Calculations have been performed to determine the required torque switch setting

5.5 Administrative Controls

It was found that the licensee's administrative program, provides for adequate control of all maintenance, testing, and surveillance activities relevant to the plant's PIVs. Maintenance histories are entered into and tracked on a database called the Production Maintenance Management System (PMMS). Test results are also stored on a database and are readily retrievable. Maintenance and testing procedures are generally well written. The licensee is in approximately the fifteenth month of a 2-year procedure rewrite program which is scheduled to be completed by January 1990. Excellent coordination of activities is provided to the plant staff by the governing procedures which are in place and functioning. Examples include ENG 1.7-55, "Documentation and Evaluation of Inservice Valves Testing", ACP 1.2-11.3, "Retest/Functional Verification, and PMP 9.5-0, "Maintenance Department Preventive Maintenance Program". There is also a procedures writing guide which standardizes the format of newly written procedures and provides guidance to the writer.

The plant staff provides good control of the hardware modification and review process. Plant Operations Review Committee (PORC) meetings are run smoothly and safety evaluation written to support the 10 CFR 50.59 review process are well written. The team attended the July 26, 1989 PORC meeting which discussed and approved a modification which installed a HPI mini flow recirculation line modification. Both design and review of this modification were technically sound. Documentation of the modification was thorough. The PORC meeting was well run.

Plant housekeeping was generally good. Areas visited by the team included portions of the primary auxiliary building, reactor containment building, turbine building, and plant exterior. All areas were clean and free of debris. Members of the team made at-power containment entries on July 26, 1989 and again on July 31, 1989. It is noteable that during the entries, respirators were not required to be worn due to low airborne levels. Also, the team received minimal exposure while inside containment due to the expert briefing prior to and accompaniment during the tours by members of the plant staff from both the Health Physics and Operations Departments.

During the conduct of the audit the licensee provided excellent cooperation and support to the team members. All concerns of the team were addressed promptly by the licensee. In several cases discrepancies brought to the attention of the licensee by the team, such as types in procedures and program documents, etc..., received attention and disposition from the licensee within one day.

5.6 Probabilistic Risk Assessment Program

A probabilistic risk assessment (PRA) program can greatly enhance the effectiveness of the engineering design process and operational readiness. It can determine the effect of the various engineering options and changes and can support in establishing certain test and maintenance programs.

The engineering staff of the licensee gave a general presentation about the capabilities, activities and present initiatives of the PRA program.

In general, the program is used to support engineering activities in a number of different areas. The most important ones are the following:

- o Evaluation of engineering design changes and their effects on plant safety as measured by the PRA analyses.
- o Evaluation of various proposed engineering options based on the safety effectiveness, primarily to improve plant safety.
- o Plant PRA model development and maintenance.
- o Evaluation of the various changes related to the Technical Specification and the various test and maintenance schedules.

The licensee demonstrated some of the capabilities of the program which included PRA model development and evaluation of various design changes through PRA calculations.

The demonstrations have indicated a very well developed computing and modeling capability using modern computer programs, remote work-station operations and knowledgeable engineering and operational personnel.

The PRA group maintains an up-to-date PRA model of the plant that simplifies the evaluation of any proposed design changes regarding its safety impact. The incorporation of the PRA in the general design process was also demonstrated including well integrated interactions with plant operational and other support engineering staff.

The PRA evaluation of the specific ISLOCA scenarios was also presented in detail and discussions about the technical aspects of the modeling were also held. In general, the PRA modeling and analysis of the ISLOCA seem to demonstrate good understanding of these events. The results of the analytical program indicates the importance of certain systems especially the low pressure injection/core deluge system.

The mechanical and hardware related failure aspects of the components on the isolation boundary are well developed and properly modelled. With the exception of the human actions related to the isolation of an ISLOCA, the event is assumed to directly lead to core damage. This is a rather conservative assumption since some of the ECCS equipment may be available to mitigate the ISLOCA event.

In this sense, the core damage frequency as predicted by the present analysis may be overpredicted. The development and inclusion of various potential operator actions would be very useful, especially, since some of the emergency procedures already include certain limited actions to mitigate or terminate an ISLOCA event. The full development of such an ISLOCA model could also include considerations of the various scenarios, affected equipment, their relations and possible mitigating options.

The human actions related to the isolation of an ISLOCA was developed as a screening function. This may have to be further analyzed, especially for the most dominating core deluge lines. These interfacing lines have one high pressure rated MOV, SI-MOV-871A and B, and in addition a low pressure rated MOV, SI-MOV-873. The actual value of the human isolation error, after an ISLOCA event, is dependent not simply on operator recognition only, but also the physical conditions of these valves due to the severity of the accident.

The PRA calculations were reviewed to a limited extent based on the preliminary documentation available and a number of minor comments were made:

o The core deluge lines have an MOV and check valve in series. Only catastrophic failure mode for the check valve is included. There are other potentially significant failure modes such as leakage and more importantly demand failure (valve fails to reseat). This latter one is important, since the check valves are opened once a month for operational testing (SUR 5.1-4 "Core Cooling Systems Hot Operational Test").

5.7 Engineering Support

Haddam Neck had received engineering support from Northeast Utilities Service Company which involved the generation of calculation No. 89-V-1132 GP, "Event 'V' Pressure Considerations for HPSI, LPSI, and RHR", dated July 18, 1989. The calculation evaluated the effects of increased static pressures from an Event 'V' (RCS pressures) for the following lines at the plant:

8"-SI-1501R-7 (HPSI)	6"-SI-1501R-5(HPSI)
8"-AC-601R-415 (RHR)	10"-AC-601R-415 (RHR)

The calculation concluded that the SI-1501R lines nominal wall thickness is acceptable for the increased pressure described in Event 'V'. With respect to the AC-601 lines, this piping significantly exceeds Code requirements for minimum wall thickness.

The NRC inspector reviewed the calculation and determined that it was accurate, clear, and technically adequate and concurs with the conclusions that the HPSI lines will withstand a static RCS pressure while the LPSI and RHR lines will experience stresses which will exceed Code allowables.

6.0 HUMAN PERFORMANCE

6.1 HUMAN FACTORS ENGINEERING

The objective of the human factors engineering audit of Haddam Neck was to identify and evaluate human factors aspects of the design, procedures, training, operation and maintenance of Haddam Neck which influence operator performance relative to initiation, detection, diagnosis, mitigation and recovery from an ISLOCA. The audit focused on four broad areas: man-machine interface, operational and maintenance procedures, communications and training. Details of the audit approach in each area and the results of the audit are discussed in the following sections.

6.1.1 Man-Machine Interface (MMI)

The MMI assessment of Haddam Neck was primarily on the main control room, such as control panel layout, control-display integration, and control and display design, which would influence operator performance during an ISLOCA event. MMI was evaluated with respect to standard human factors design guidelines as contained in NRC NUREG-0700, "Guidelines for Control Room Design Reviews, 1981". Within the MMI assessment, special emphasis was placed on the availability and quality of information necessary to detect and diagnose an ISLOCA precursor situation. In addition to the main control room, the MMI assessment examined selected aspects of local panels and components located in the PAB and inside the reactor containment. The methodology employed during this portion of the audit involved direct observation of equipment and facilities and interviews and discussions with plant personnel. No quantitative measures of the physical characteristics of environments or equipment were made; however, subjective assessments of these aspects of the MMI were performed.

6.1.1.1 High Pressure Injection system pressure, temperature and flow indication

During a postulated ISLOCA event in the HPI injection line involving failure of check valve SI-CV-862 in combination with a failure of the in-line stop valve, SI-MOV-861, the HPSI injection line could be over-pressurized. Although the HPI relief valve, SI-RV-871, would relieve pressure on the line by discharging to the reactor water storage tank (RWST), detection of the ISLOCA precursor condition would be delayed due to the lack of HPI pressure, temperature and flow indication in the control room. The only indication of HPI system operation available to an operator in the control room was HPI pump motor amps, which were used to infer pump discharge pressure. The plausibility of this scenario derives from maintenance history on the SI-MOV-861B and SI-CV-862B which failed leak rate tests in the past (See section 7.1 of this report for additional details on the history of these valves).

It should be noted that the existing emergency operating procedures for a reactor trip/safety injection, E.0, did not include reference to the RWST level as an indication of a LOCA outside containment (see section 6.1.2.1 for additional details).

The lack of direct indication of HPI pressure, temperature and flow is not consistent with the intent of NUREG-0700, paragraph 6.1.1.1, "Accessibility of Instrumentation/Equipment".

6.1.1.2 Refueling water storage tank (RWST) low-low level alarm

One of the potential consequences of the postulated ISLOCA scenarios is the unavailability of water for the recirculation mode of core cooling. This loss of water can be attributed to the depletion of RWST inventory. Since the LPSI pumps would begin to cavitate at approximately 67,000 gallons and HPSI pumps would begin to cavitate at approximately 43,000 gallons, a low-low RWST level alarm could be provided to direct the operator's attention to the loss of RWST inventory. There was no alarm for RWST low-low level in the control room. There were, however, two annunciators for RWST level: a High/Low Level alarm (250,000 and 230,000 gallons, respectively) and Switchover to RHR recirculation alarm (130,000 gallons).

It should be noted that the existing operating procedures for transferring RHR supply suction from the RWST to RHR recirculation, "ES-1.3", step 12a, directs the operator to verify that containment water level of 2.25 feet exists before starting the RHR pumps; however, there is no "Response Not Obtained" entry for that step (see section 6.1.2.1 (iv). In addition, the display used to monitor containment level has a minimum indication by design of 1.5 feet. (see sections 6.1.1.9 and 6.1.2.1, (iii)).

The absence of an alarm for RWST low-low level in conjunction with related procedural and display deficiencies, increases the likelihood that an operator may fail to correctly detect and diagnose the ISLOCA condition.

6.1.1.3 RWST level trend recorder

RWST level has been determined to be important for early detection/diagnosis of some ISLOCA precursor conditions such as the venting of an ECCS relief valve to the RWST. The only RWST level trend recorder was located on the "Green Panel" in the PAB.

In addition, the trend recorder was located below the normal line of sight for a standing operator (approximately 48" off the floor). Also, the label for the recorder was obscured by the recorder housing. This was a particular problem since LR 1806 was a three pen recorder, with the other two pens recording another tank's levels. Although the first line of the label for this recorder clearly designated one of the channels as RWST level, this line may not be visible to a large majority of operators. This design may increase the likelihood that an operator will be delayed in locating display of RWST level trend, with a consequent delay in diagnosing a ISLOCA condition.

This design is not consistent with the intent of NUREG-0700, paragraph 6.1.2.2, "Stand Up Console Dimensions".

6.1.1.4. RWST high temperature alarm

During an extended ISLOCA involving venting of an ECCS relief valve to the RWST, it is conceivable that RWST temperature could increase, with a subsequent alarm condition. The "RWST High Temperature Alarm" was located on the "Green Panel" in the PAB. RWST High Temp was annunciated in control room as simply "PAB Alarm;" therefore, to determine the precise nature of the alarm, the control room operator must instruct an AO to determine the cause of the alarm and report back.

6.1.1.5. RWST Level alarm multiple input signals

The RWST level alarm in the control room receives both high and low signals. During an ISLOCA situation, RWST level could be a critical indicator of plant status. Viewed in the context of human factors problems with related procedures (see section 6.1.2.1), the inherently ambiguous meaning of the multi-input RWST level alarm increases the probability that the operator will fail to detect or diagnose the ISLOCA condition.

NUREG-0700, paragraph 6.3.1.2 C(1), recommended that multi-input annunciators should be avoided.

6.1.1.6. Containment water level indication

Control room trend recorders for containment water level, LR 1810A on panel EE and LR 1810B on panel FF indicated a minimum value of 1.5 feet. As a result, operators could not determine the actual water level if it is less than 1.5 feet. There was an additional trend recorder for containment sump water level on main control panel section E; however, the instrument that provides input to the recorder was not environmentally qualified and therefore, cannot be considered reliable during accident conditions in the containment. Procedure ES-1.3, "Transfer to RHR Recirculation, (Rev 7, 7/7/89) step 3.a, directs the operator to verify that containment water level is greater than 2.25 feet prior to starting the RHR pumps; however, there is not a "Response Not Obtained" action for this step (see section 6.1.2.1, (iii). During discussions with the control room operators, the operators stated that they considered the LR 1810A/B recorders to be "fairly inaccurate."

The safety significance of the combination of limited display range, procedures and operator assumptions regarding the containment water level display accuracy, is that operators may fail to detect the absence of water in the containment. This situation has two distinct consequences with respect to operator performance during an ISLOCA. First, since the absence of water in containment should be a primary indication of a potential ISLOCA, the operator may be delayed in detecting/diagnosing the condition. Second, if the operator follows the current procedure verbatim, he may attempt to start the RHR pumps to initiate RHR recirculation without the necessary supply of water. This could result in damage to the RHR pumps and subsequent loss of a core cooling train.

6.1.1.7 <u>Containment isolation valve status</u>

ECCS MOV position indication are based on a signal from limit switches on valve stem.

This design was consistent with NUREG-0700, paragraph 6.5.1.1. e, "Demand Versus Status Information", and was considered to reduce the likelihood that an invalid indication of valve position will be displayed. EOP E-0, "Reactor Trip or Safety Injection" (Rev 6, 7-7-89), step 17, directs the operator to verify that forty four valves must be verified to be closed. Of those 44 valves, 22 have their position indications displayed in the main control room, 12 have only local position indication displayed in the PAB, and 10 must be verified closed at the valve itself locally. This design may increase the operator's workload on and imposes significant requirements for communications and coordination between control room operators and AOs in the plant.

6.1.1.8 <u>Inadvertant RHR valve Actuation</u>

The RHR outboard stop valves, RH-MOV-781 and 803 had been designed to minimize the probability of inadvertent actuation. This design included: 1) pressure interlocks which prevent the valves from being opened when RCS pressure is greater than 400 psig; 2) key-operated control switches on the main control board; 3) "locked open" disconnect switches which remove power from the control circuit; and, 4) annunciators which inform the operators that the valves are opening or closing.

These design practices were consistent with NUREG-0700, paragraph 6.4.1.2, "Prevention of Accidental Activation", and were considered to virtually eliminate the possibility of inadvertent actuation of the RHR outboard stop valves.

6.1.1.9 Main control board component arrangement

The arrangement of components on the C section of the main control board was not consistent. The controls for SI-MOV -861 and 871, the HPSI A and B pumps and the LPSI A and B pumps all had a vertical arrangement (i.e, form a vertical line with A at the upper position). In contrast, the RHR A and B pumps had a horizontal arrangement, with the A pump on the left. This design arrangement may increase the. probability that the operator will commit an error in locating a control when working across the RHR and SI systems.

This design was considered to be a departure from the intent of NUREG-0700, Paragraph 6.8.2.3, "Layout Consistency".

6.1.1.10 Control room component integration

EOP E-0, "Rector Trip or Safety Injection" (Rev 6 7/7/89), step 17b, directs the the operator to verify that both let down trip valves are closed. However, LD-TV-230's control was not co-located with other letdown valve controls. LD-TV-230's control was located on the Post Accident Monitoring (PAM) "EE"panel which was located opposite to the main control panels; the controls for the LD-MOV-200 valves was located on main control board MCB-C. This may require that the operator at the main control board enlist the aid of another operator at the PAM panel. This situation may unnecessarily contribute to operator workload, and increases the probability that an operator error will precipitate or exacerbate an ISLOCA event requiring the use of EOP E-0.

This valve control location design is not consistent with the intent of NUREG-0700, 6.8.1.1, "Assigning panel contents".

6.1.1.11 Control Room Labeling

Control room labeling for the safety injection WL relay controls did not correspond to applicable operating procedure terminology and standard operator designation for the safety injection WL relays. Operating procedures and operator terminology refer these controls as the "Safety Injection WL Relays;" however, the control panel labeling designates these controls as "Core Cooling A&B". This design increases the probability that the operator may be delayed in initiating ISLOCA mitigative action should these controls require manipulation.

This design is not consistent with the intent of NUREG-0700, paragraph 6.6.3.3 (c), "Consistency with Procedures".

6.1.1.12 Component Labeling

Existing plant component labeling for an ISLOCA event were found to be generally adequate. During walkdowns of ECCS systems, the team noted that most component labels were attached using a metal loop and not mounted directly on the equipment. This configuration may allow the label face to be turned away from the operator's working position. This problem was noted on several ECCS valves within containment, in which case the operator would have to reach over or around other equipment to manipulate the label in order to read it. This design may increases the probability that an operator will be delayed in locating an equipment item and unnecessarily cause the operator to spend additional time in a radiation area.

This design was a departure from the intent of NUREG-0700, paragraph 6.6.1.1, "Need for Labeling".

The team noted that the plant labeling program in place during the audit was well structured and controlled by procedure. Labels were found on all plant components reviewed by the team. The new labels were clearly written, easily read, most were visible, had adequate contrast between lettering and background, were properly located generally and were consistent with plant drawings and procedures.

6.1.2 Operating and Maintenance Procedures

Haddam Neck normal, abnormal and emergency operating procedures were reviewed relative to postulated ISLOCA events. The procedures were reviewed to determine that verbatim execution of the procedures would assist or impede the operator in detecting, diagnosing , mitigating and recovering from an ISLOCA event. The electrical and mechanical ECCS motor operated valve maintenance procedures were reviewed to determine if they contained adequate information to assure that the objective of each task was achieved and to ensure adequate valve maintenance and thus their reliability. The focus of the review was on 1) the accuracy, completeness and clarity of the procedures; 2) the feasibility of the procedures given expected plant conditions and available personnel; and 3) the extent to which the procedures specifically address the operational requirements and concerns of an ISLOCA event. In addition to the content review, the process by which operating procedures were developed and validated, including the emergency operating procedures (EOP) writer's guide, was assessed.

6.1.2.1 Emergency Operating Procedures (EOP)

The Haddem Neck ISLOCA emergency operating procedures were reviewed and identified the following findings were identified:

(i) EOP E.0, "Reactor Trip - Safety Injection" (Rev 6, 7-7-89), step 29, did not include a reference to check for a change in RWST level as an indication of a possible LOCA outside containment. The existing procedure lists only PAB radiation level as a possible indication. RWST level indication could increase the likelihood that the operator will correctly diagnose the ISLOCA condition and branch to EOP ECA 1.2, "LOCA outside containment".

(ii) EOP ECA 1.2, "LOCA outside containment" (Rev 1, 6/3/88), step 1.a, directs the operator to verify that RHR loop 1 suction valves RH-MOV-780 and 781 are closed; step 1.b directs the operator to verify that RHR loop 2 suction valves RH-MOV-803 and 804 are closed. The "Response Not Obtained" entries for these steps direct the operator to manually close the valves, and, if necessary to close the valves locally. Given that valves RH-MOV-780 and 804 were located close to the reactor coolant loop RHR penetrations, operator access to them may be difficult.

(iii) EOP ECA 1.2, "LOCA outside containment" (Rev 1, 6/3/88), step 1.c, directs the operator to cycle the core deluge valves SI-MOV-871 A and B and monitor for an reactor coolant system pressure increase. Step 1.c did not contain a "Response Not Obtained" entry. The procedure could inadvertently allow the operator to establish a ISLOCA precursor by cycling the MOV and leaving it in the open position. This missing step should warn the operator of the potential for an ISLOCA condition if RCS pressure increases after the subject MOVs are closed, and instruct the operator not to re-open the MOV.

(iv) EOP ES-1.3, "Transfer to RHR Recirculation" (Rev 7, 7/7/89) step 3.a, directs the operator to determine if the containment water level is greater than 2.25 feet. This step did not include a "Response Not Obtained" entry. The lack of the expected water level in containment could be considered a possible indication of a ISLOCA. Further, since this level is required to ensure that RHR pumps do not cavitate, the procedure did not caution the operator not to continue until the necessary level has been confirmed.

(v) EOP E.1, "Loss of Reactor or Secondary Coolant" (Rev 5, 7/7/89) step 12.a, directs the operator to verify RHR recirculation capability and did not include an action/precaution to assure that containment sump water level is greater than 2.25 feet. Since this level is required to ensure that RHR pumps do not cavitate, the operator should be instructed to confirm the necessary level before starting the pumps.

(vi) The Hadoem Neck EOPs contain numerous requirements for AOs to support control room operators during emergency conditions. For example, in an ISLOCA event involving SI initiation, failure of one or more of the RHR loop isolation valves and an RWST alarm in the PAB, it is conceivable that four AOs would be required to perform parts of applicable the EOPs: one to enter containment to locally close the RHR valves; one to determine the cause of the PAB RWST alarm; one to check remote SI valve position indications in the PAB; another to check local SI valve indications. To implement EOP, ECA 1.2, "LOCA Outside Containment", an additional operator may be required to verify approximately 10 other flow paths/valve positions, some of which require entering a pipe chase that may contain conditions that would not be conducive to effective human performance: access space is insufficient, lighting is inadequate, labeling is poor, temperature and humidity may be excessive, and respirator or air pack may be required. Given the complexity of the coordination and communication requirements inherent in this scenario, there may be a high likelihood of operator error.

Collectively, these deficiencies could increase the probability that the operator will fail to detect or diagnose an ISLOCA condition, or will fail to initiate mitigative action in a timely manner.

6.1.2.2 Procedure Development Program

The Haddam Neck procedures development/revision program included iterative reviews by operations and training personnel. This would reduce the likelihood of errors in procedures and increase operator awareness of pending procedural changes.

Maintenance procedures used by electrical and mechanical personnel with the repair of RHR and SI MOV's were quite good. the procedures were well written using good human factors practices and guidelines. The procedure's action steps were written and arranged to enhance user comprehensibility. Caution and warning statements were highlighted by including boxing around them and did not contain action steps. Illustrations included in these procedures were clear, accurate and were generally located with applicable action steps. The vocabulary and abbreviations, acronyms and symbols used were consistent with plant standards. The action steps were sequenced properly and the verification steps assured that the objective of the task was achieved. These characteristics combined to form a useful tool to assure accurate valve maintenance.

6.1.3 Communications

Communications were evaluated in terms of training and protocols for exchange of information between control room operators and auxiliary operators (AOs).

The emphasis was placed on the control room operators/AO communications during an ISLOCA event. In addition to communications practices, the equipment used for communications was evaluated relative to sound quality and location.

Communication capability between the control room and containment was determined to be limited to the in-plant telephone system. Although the quality of the transmission was good, the location of the telephone within containment, relative to the location of the SI valves, may require the operator to repeatedly transit between the valve location and the telephone which was a distance of approximately 50 feet. This situation may unnecessarily contribute to operator workload and increases the probability of error. It should be noted that the environmental conditions within containment; high temperature, high noise levels and poor lighting, coupled with the psychological stress associated with making an entry into containment during an event also contributes to operator burden and attendant error probability.

6.1.4 Training

The Haddam Neck operator training program and the electrical and mechanical maintenance training programs were evaluated to determine the extent to which current training practices prepare an operator to detect, diagnose, mitigate and recover from postulated ISLOCA events and if maintenance personnel were prepared to adequately maintain and repair essential ECCS components to assure their reliable operation. This evaluation consisted of interviews with operations, maintenance and training personnel and review of applicable training materials.

6.1.4.1. Simulator Training

During interviews with control room operators, it was noted that they had not received simulator training on ISLOCA detection, mitigation or recovery in general, or on procedure ECA 1.2, "LOCA Outside Containment". The current simulator configuration does not have the capability to simulate a failed ECCS check valve, and was not capable of simulating most postulated ISLOCA events. In addition, there were currently no scenarios in the simulator training program which contain reactor coolant leaks which would be large enough to require operators to use procedure ECA 1.2.

6.1.4.2. Auxiliary operator and maintenance training

Auxiliary operators (AO) received a minimum of one day of simulator training per training cycle and this training emphasizes team coordination and communications between control room operators and AOs during simulated emergency conditions. This is a positive feature which may reduce the likelihood of communications problems between control room operators and AOs during an ISLOCA event.

The training programs for maintenance personnel responsible for the repair of ECCS components critical to ISLOCA prevention and mitigation was also reviewed. The training facility and program were adequate, and the administrative control procedures were adequately implemented. The knowledge and experience of the mechanical and electrical maintenance was quite good. They possess requisite experience and qualifications that were commensurate with their responsibilities and assigned functions. They were sensitized to the importance of reactor systems and equipment and understood how their performance was reflected in plant reliability and safety.

6.2 Human Reliability Evaluation

6.2.1 The Identification of Human Actions and Errors

In order to identify potential human errors, human actions associated with ISLOCA were first identified. This was accomplished by noting the human actions associated with interfacing systems for normal and off-normal situations. The human actions identified were evaluated on the basis of the postulated ISLOCA scenarios.

There were four scenarios associated with LPI/Core deluge, three scenarios each associated with RHR injection, and four scenarios associated with HPI. A list of the human actions and potential for errors associated with each of the systems and scenarios is presented in Tables 6.2 through 6.4. Four classes of errors were identified. The definition of each error class used to support human reliability evaluation of ISLOCA is as follows:

(1) human initiators - operator opens a MOV, leaky check valve already exists

(2) human initiators which are immediate precursors - improperly executed valve line-up followed by leaky check valve

(3) human actions during repair which can compromise equipment - installing the wrong seals or miswiring

(4) human actions related to mitigation or aggravation - failure to detect an IS LOCA situation or improper diagnosis

6.2.2 Performance Shaping Factors

6.2.2.1 Introduction and Background

Human performance in systems is subject to influence from a variety of sources. These sources may be part of the task itself, part of the environment, or part of the history, make up or physical limitations of the persons themselves. For example, a person's experience or lack thereof may either support or detract from their overall task performance. Similarly, the pacing of the task may facilitate or reduce performance. Adequacy of instrumentation or system feedback may do the same. These sources of variation in performance are referred to by human reliability analysts as performance shaping factors (PSFs).

As part of this evaluation, relevant PSFs for those human actions identified in step 6.2.1 were determined. These PSFs were then reviewed for either positive or negative influence on human reliability. The PSFs were rated as either a "+" or "-" for each action identified in the scenarios described above. This was accomplished by combining interviews with plant personnel, observations, inspection team findings, walkdowns, and reviews made against standardized human factors criteria contained in NUREG/CR-4835. A list of these PSFs with their definitions is provided below.

TABLE 6.1

LIST OF PERFORMACNE SHAPING FACTORS

Training -	Training: either classroom, on the job, or simulator supports ISLOCA detection, response, and diagnosis.
Feedback -	Indications are available and lag time is not detrimental to personnel performance.
Crew Size -	Crew size does not lend itself to either overmanning or under manning.
Procedures -	Procedures are accurate, legible, and complete with an acceptable format.
Workload -	Physical and mental workload do not diminish the crew's capacity to respond.
Workshift -	Positive unless errors occurring after personnel have performed a double shift, or are working unusually fast rotations (e.g., in high temperatures, high radiation environments).
Supervision -	Is it present, are the supervisors well trained, and to the extent it is possible to determine what is the quality of supervision.
Experience -	Refers to the the experience level of the crew in near ISLOCA situations and in simulator training. It is assumed that experience will tend to support adequate performance for the particular IS LOCA sequence.
Environment -	Refers to light, humidity, radiation, noise level as it impacts performance. For example, some valves may be inaccessible not due to their height but rather to their location within a radiation environment.
Operator burden -	A combination of workload, time available and the complexity of the task.
Communication -	Physical systems as well as interpersonal communications (e.g., misunderstandings due to dialect or failure to institute verification procedures such as repeating back a phrase to a speaker).

Task
Location - Local (at the component) and remote (control from the control room). Positive ratings indicate the action is remotable, negative that personnel must be sent into radiation or steam environments.

Plant sequence
difficulty - Positive refers to instances for which there are procedures and plant phenomenology is understood by operations or maintenance personnel.

Cognitive
complexity - Positive indicates that symptoms do not mask potential plant states and that personnel have rules they can follow and do not have to work within the knowledge base domain.

Machine paced
task - Positive indicates an optimal level where pacing is neither too slow nor too fast. Particularly relevant where tasks are overlapping and the first started may continue well beyond completion of the second or third task.

Man Machine
Interface
(MMI) Refers to ergonomics (reach, legibility, design for maintenance, accessibility, adequacy of information presented, controls/input devices.

Stress -- Mismatch between perceived task requirements and perceived ability to respond, can result from an intellectual demand or physical demand (too many tasks having to be performed within a short period of time) or by the consequence of the actions needing to be performed (where poor performance might lead to safety ramifications, loss of plant revenue, or loss of operating license).

Circumvention -
An action taken outside of an existing procedure in order to meet an alternate or higher order safety goal. A positive rating indicates there is little chance of circumvention occurring.

6.2.2.2 Evaluation Matrix

Results of the evaluation of performance shaping factors for all actions and scenarios were placed in a PSF matrix which is reproduced as Table 6.5. Human actions (designated A through E) are nested within systems (LPI, HPI, and RHR) which are, in turn, grouped by scenario. The actions, A through E, correspond to those actions presented earlier in Tables 6.2 through 6.4. For simplicity, similar scenarios have been linked together. For example, in column 2, LPI/CD and HPI 1 actions are so similar that they have been grouped together. Likewise, actions required in LPI/CD2, RHR 1, and HPI 2 are similar. Results of the PSF review are similar and have been grouped together for the reader's review. Definitions for the group of PSFs selected for presentation in Table 6.5 may be found in Section 6.2.2.1. As presented, "+" indicates a positive influence on performance, "-" indicates a negative influence and a "blank" indicates either a neutral influence or that the PSF applied to some actions but was not relevant for others. The PSF "circumvention" defined as "actions outside of or in conflict in the procedures" is included because of suspected importance in certain ISLOCA situations.

In Table 6.5, the human actions and the corresponding event scenarios are grouped together in accordance with their similarities, and Table 6.2 is organized by the potential human actions/errors. For example, "A "through" E" in Table 6.2 refers to specific human actions. It is interesting to see that the diagnosis aspects of human actions/errors in Table 6.2 are 1.C, 2.E, 3.C, or 4.B.

TABLE 6.2 - Human Actions/Errors Identified for Low Pressure Injection/Core Deluge (LPI/CD) Systems

1. - Spurious SI signal
   - MOV 871A and 871B open on SI signal
   - A - Operator fails to diagnose spurious SI, and
   - B - Operator fails to close MOVs, (would not close until called out for in procedures ECA 1.2)
   - One or both CVs (872A and 872B) fails
   - C - Operator fails to detect ISLOCA situation

   Note: A spurious SI contact closure (i.e., short) could open an MOV which could not then be closed by the operator.
   871A and 871B (Core Deluge) are not accessable for manual operation.

2. A - Operation or Maintenance fails to return MOVs 871A and 871B to closed position following test or maintenance

   B - Operator fails to follow valve lineup procedure

   C - Operator fails to notice open MOVs during shiftly CR check, or

   D - Operator fails to initiate/perform shiftly CR check

   - CV (872A and/or 872B fails on same train(s) as open MOV(s)

   E - Operator fails to detect ISLOCA situation

   Note: For Scenarios 1, 2, 3, and 4 CD line can be isolated by closing MOV-873 isolation valve (not energized till after next outage, is accessible for local manual operation).

3. A - Operator mistakenly opens MOV-871A and/or MOV 871B (CD)

   - CV (872A and/or 872B) fails on same train(s) as open MOV

   B - Operator fails to detect open MOV(s) during shift turnover

   C - Operator fails to detect ISLOCA situation

   *Large leak followed by SI

   *Small leak followed by a drop in the VCT

4. - Valve stems separation (MOV-871A or 871B) near beginning of fuel cycle; valve is in open position but CR indication is closed

   A - Operations or Maintenance fail to identify valve stem separation during post maintenance/pre startup testing

   - CV (872A and/or 872B fails on same train(s) as open MOV

   B - Operator fails to detect IS LOCA situation

TABLE 6.3 - Human Actions/Errors Identified for the Residual Heat Removal (RHR) System

RHR Injection

1. A - Operations or Maintenance fails to return inboard isolation (MOV 804) to closed position following test or maintenance (outboard isolation MOV-803 is pressure interlocked, locked out at breaker, and key locked at control switch)
   B - Operator fails to follow valve lineup procedure
   C - Operator fails to notice open MOV during shiftly CR board check, or
   D - Operator fails to initiate/perform shiftly CR board check
   - MOV 803 fails
   E - Operator fails to detect ISLOCA situation

2. - Valve stem separation (MOV-804 or MOV-803), valve in open position but CR indication is closed
   A - Operations or Maintenance fail to identify valve stem separation during testing or normal operations
   - Failure of remaining MOV
   B - Operator fails to detect ISLOCA situation

3. A - Operator mistakenly opens MOV-804
   B - Operator fails to detect open MOV during shiftly CR board check or because didn't note expected effect of MOV initially intended to open
   - MOV-803 fails
   C - Operator fails to detect ISLOCA situation

RHR suction

- RHR suction isolation valves, MOV-730 (inboard) and MOV-731 (outboard), are modeled as the injection valves above

TABLE 6.4 - Human Actions/Errors Identified for the High Pressure Injection (HPI) System

HPI

1. - Spurious SI sig. opens MOV-861 A, B, C + D (isolation)
   A - Operator fails to diagnose spurious SI, and
   B - Operator fails to close MOVs, and
   - One or more CV 862 A, B, C & D fail
   C - Operator fails to detect ISLOCA situation

   Note: A spurious SI contact closure (i.e., short) could open an MOV which could not then be closed by the operator.

2. A - Operations or Maintenance fails to return one or more MOV(s) to closed position following test or maintenance
   B - Operator fails to follow valve lineup procedure
   C - Operator fails to notice open MOV(s) during shiftly CR board check, or
   D - Operator fails to initiate/perform shiftly CR board check
   - CV(s) fails on same train(s) as open MOV(s)
   - Operator fails to detect ISLOCA situation

3. A - Operator mistakenly opens one or more MOVs
   B - Operator fails to detect open MOV(s) during shiftly CR board check or because didn't note expected effect of MOV(s) initially intended to open
   - CV(s) fail on same train(s) as open MOV(s)
   C - Operator fails to detect ISLOCA situation

4. - Valve stem separation (MOV 861 A, B, C & D) at beginning of fuel cycle; valve(s) is in open position but CR indication is closed
   A - Operations or Maintenance fail to identify valve stem separation during post maintenance/prestartup testing
   B - Operator fails to detect ISLOCA situation

   Note: Human actions involving valve mispositioning may be influenced by faulty valve position indications (i.e., failure of limit switch mechanisms or maintenance errors in wiring).

TABLE 6.5 PREFORMANCE SHAPING FACTORS ASSESSMENT MATRIX FOR ISLOCA

PERFORMANCE SHAPING FACTOR (PSF)	SCENARIO HUMAN ACTION												
	LPI/CD 1			LPI/CD 2					LPI/CD 3			LPI/CD4	
				RHR 1					RHR 3			RHR 2	
	HPI 1			HPI 2					HPI3			HPI4	
	A	B	C	A	B	C	D	E	A	B	C	A	B
TRAINING	+	-	-	+	+	+	+	-	+	+	-		-
FEEDBACK	-	-	-	+	+	+	+	-		+	-	-	-
CREW SIZE													
PROCEDURES	+	+	-	+	+	+	+	-		+	-		-
WORK LOAD	-	-	-	-	-	+	+	-		+			
WORK SHIFT	-			-	-	-	-			-			
SUPERVISION	+	+		+	+	+	+			+			
EXPERIENCE	+		-	+	+	+	+	-		+	-	-	-
ENVIRONMENT		-	-	-	+	+	+	-		+	-	-	-
OPERATOR BURDEN	-	-	-	-	-	+	+	-		+		-	
COMMUNICATION	+	+	+	+	+	+	+	+		+	+		+
TASK LOAD	-	-	-	-	-	+	+	-		+	-	-	-
PLANT SEQUENCE DIFFICULTY	+		-	+	+	+	+	-		+	-		-
COGNITIVE COMPLEXITY	+		-	+	+	+	+	-		+	-		-
MACHINE PACED TASK	-	-	-	+	+	+	+	-			-		-
MAN-MACHINE INTERFACE	+	+		+	+	+	+			+		-	
STRESS	-	-	-										
CIRCUMVENTION*													

*Data not available

6.2.3 Preliminary Findings

Although time constraints precluded performance of an in-depth quantitative analysis, the qualitative analyses described in earlier sections suggests the following:

- PSFs are generally positive for ISLOCA relevant actions that typically appear in other plant evolutions (not related to ISLOCA).
- PSFs are generally negative for detection/diagnosis of ISLOCA situations, particularly for HPI.
- The PSF "psychological stress" has potential negative influence on human reliability following a safety injection signal, or in off-normal situations.
- Personnel all appear to be sensitive to the importance of good communication. The PSF for communication is therefore generally positive. However, there are not enough physical lines of communication into the control room under certain emergency situations.
- This human reliability evaluation approach provides insight into prevention/mitigation of ISLOCA (e.g., identifying important actions and pinpointing PSFs which can reasonably be assumed to have a negative influence on reliability).

6.2.4 Review of CY HRA in PRA for ISLOCA

The CY HRA reviewed in support of this ISLOCA effort was valuable for the insight. It modeled human actions for a number of important plant systems and operational sequences. It was limited, however, as a resource in terms of evaluation of performance shaping factors. Table 6.6 presents a comparison of the attributes of a state-of-the-art HRA with the HRA performed previously by CY.

TABLE 6.6 - Comparison of HRA Approaches

	State-of-the-art of HRA		CY ISLOCA HRA
o	Performed as joint effort by HRA and system analysts	o	Performed by system analyst
o	Standardized stepwise approach such as SHARP to ensure completeness and traceability. SHARP steps: 1) Definition 2) Screening 3) Breakdown 4) Representation 5) Impact Assessment 6) Quantification 7) Documentation	1)	System analysis to identify human actions 2) Screening and limited THERP application for quantification of HEPS
		3)	Incorporation of screening values and HEPs in PRA probability statements
		4)	Documentation
o	Conservative screening values used for human actions "across the board" to identify actions important for more detailed analysis in PRA context	o	Screening values used only for cognitive type human actions
o	Quantification through estimation of HEPs for identified important actions human actions by careful, complete, detailed application of an appropriate HRA technique (i.e., THERP, HCR, SLIM-MADD, etc.)	o	Quantification through use of screening values for cognitive and limited THERP application for other actions.
o	HEP estimates include assessment and consideration for: 1) dependence between actions and between humans, 2) recovery factors, and 3) relevant performance shaping factors	o	Dependence not assessed; limited consideration of recovery factors and performance shaping factors

7.1 Potential Precursor

7.1.1 Leakage Across Pressure Isolation Valves in the HPI Header

The audit team reviewed the tabulated surveillance test results of the valves classified as PIVs in the RHR, HPI, and the LPI systems. The review revealed that both the primary and secondary PIVs in one of the four HPI headers were leaking during the same period. The review also indicated that the leakage past these PIVs was identified by the licensee and that the licensee had successfully performed corrective maintenance on these PIVs. However, the significance of simultaneous leakage in two PIVs (SI-MOV-861B and SI-CV-872B) located in the same discharge header was not recognized as a breakdown in the pressure isolation capability of these PIVs and as a potential ISLOCA precursor by the licensee.

Furthermore, detection of the above leakage could have been delayed in the event that the leakage had occurred while the reactor was operating at power. This is based on the finding that there was no pressure instrumentation on the HPI discharge header nor were there any over-pressure alarms locally or in the control room.

However, the header was equipped with a relief valve with a capacity of 33 gpm which relieves from the discharge header to the Refueling Water Storage Tank (RWST). Consequently, leakage across two PIVs has to exceed the relief valve capacity before over-pressurization of this piping was possible. Although the RWST is vented to the atmosphere and has no radiation alarm to detect for possible RCS leakage into this system, its level reading was recorded daily so that its level increase would probably be detected by the operators. Leakage across SI-MOV-861B and SI-CV-872B was found to be 1.21 gpm and 50.9 gpm, respectively, in surveillance performed on April 7, 1986 during their thirteenth outage. The acceptable leakage across both of these valves is 1 gpm.

Fortunately, both of these valve leakages had occurred during shutdown conditions because these two PIVs successfully passed surveillances performed on January 13, 1986 and the plant was shutdown for the refueling outage on January 4, 1986.

7.1.2 Leakage Across MOV in the Core Deluge Header

If the MOVs for pressure isolation in the core deluge line were leaking, it would be possible for operators to inadvertently open these valves with the reactor system at power. This would be possible because their upstream check valves, having been tested and found to have no back leakage, would act as a pressure boundary.

The core deluge discharge MOVs cannot be leak tested due to their proximity to these check valves, and it is reasonable to expect some leakage from valves which have been in service for a long period of time (in this case, over twenty years). Hence, it appeared that design features which one would normally expect, i.e., those that would prevent core deluge line MOV operation with high differential pressure felt across them, may not be realized. If operator error in opening the MOV is assumed to be the initiating event, and the failure of the check valve to be the single failure, then an ISLOCA event could occur. The worst case consequences of this accident would be the rupture of RHR piping with subsequent RHR system failure. Unlike the RHR MOVs with multiple interlocks and alarms, these core deluge line MOVs have no interlocks.

7.2 ISLOCA Mitigation

Inadvertant overpressurization of the low pressure portions of systems and components interfacing with the RCS can lead to ISLOCAs with potentially significant contribution to the public risk. Previous PRA analyses have indicated that the core damage frequency contribution from ISLOCA events is relatively low. However, their contribution to the public risk could be high for the following reasons:

- o The containment would be bypassed and direct pathway to the environment may be established
- o Systems designed to mitigate loss-of-coolant type of accidents may directly or indirectly be affected causing the partial or t tal loss of the mitigation capability

The response to an ISLOCA event is determined by the duration of the accident. The primary concern is the ability to inject cooling water into the RCS to cool and remove heat from the reactor core, in both the short and long term.

1. Short term considerations

- o The location of the HPI and LPI pumps are of concern. An ISLOCA event in this open pump pit area could affect both the HPI and LPI portion of the ECCS, since physical barrier is not provided between the safety components. In a small break ISLOCA the HPI may be replaced with the charging system which is capable of providing relatively large volume (300 GPM) of high pressure injection flow.

  However, a large break ISLOCA event would definitely require the operation of the LPI pumps and on the long term may also require the HPI system. Based on these considerations, the large break ISLOCA would be the scenario most affected by the lack of physical separation in the HP/LP pump pit area.

- o The charging system is physically separated from the HPI system and is expected to maintain its integrity during all ISLOCA event with the possible exception of a large release in the PAB.

- o The emergency procedures (E-0, ECA-1.2, ECA-1.1) are designed to assist the operator during an ISLOCA emergency. The procedures are not clear with respect to the potentially affected equipment/lines and the possible loss of water inventory of the RWST through the break.

- o The accident may damage the HPI or LPI/RHR injection or suction lines causing the loss of the injected water through the break before it reaches the reactor core. This coupling is not recognized by the procedures and the operators seemed to be uncertain regarding the importance of this potential situation.

2. Long term considerations

o The primary coolant and the additional injected cooling water may accumulate in the PAB. One of the most likely locations may be in the RHR pump pit. The accumulated coolant may then be recirculated either by temporary "ad hoc" arrangements or by modifying an already existing small system used for purification purposes. At present, the plant operational staff has no procedures or any training in this area.

7.3 ISLOCA Consequences

The offsite consequences of postulated ISLOCA events depend on the break size and timing of fission product releases. If core cooling cannot be maintained, core damage occurs and the fission products are then released into the surrounding environment through the break location. The following observations were noted in this regard:

- The PAB is a relatively small, compact building with many pathways to the outside. The building is not leak tight with numerous doors leading to the plant yard. The pressure retention capability of the building is minimal due to the open or easy opening doors and the corrugated steel side and roof structure of the top floor.

- Some of the low pressure piping is inside the pipe chase that may localize small releases. However, the pipe chase communicates with the PAB through manholes and due to the small volume overpressurization is expected in the chase with consequent release to the rest of the building.

- A large release may occur inside the RHR pump pit area. This arrangement, with a potentially large accumulation of the coolant, may provide scrubbing effects of the fission product releases.

- Additional scrubbing effects can be achieved if the limited fire spray system would turn on. The fire spray system is installed at various locations of the building and actuated by heat sensitive detectors. A large primary coolant release inside the PAB would increase the temperature inside the building, but not necessarily to the level to initiate the fire spray system (set at 200° F, no manual actuation). It should be noted that the system is not very extensive and no major reduction of the fission product release is expected by its use.

8.0 Exit Meeting

The team met with the licensee representatives denoted in paragraph 1 on August 4, 1989, and summarized the purpose, scope and findings of the audit. The attendees are listed in paragraph 1 of the report details.

APPENDICES

Appendix A - Reference Documents

1. Piping and Instrumentation Diagrams

Drawing Number	Sheet	Rev.	Title
16103-26007	1 of 3	9	P&ID Reactor Coolant System Loops 1 & 2
16103-26007	2 of 3	8	P&ID Reactor Coolant System Loops 3 & 4
16103-26007	3 of 3	6	P&ID Reactor Coolant System Pressurizer
16103-26010	1 of 1	29	P&ID Safety Injection System
16103-26078	1 of 1	13	P&ID Residual Heat Removal System
16103-26018	1 of 8	11	P&ID Chemical & Volume Control - Letdown to Volume Control Tank
16103-26018	2 of 8	11	P&ID Chemical & Volume Control - Purification
16103-26018	3 of 8	14	P&ID Chemical & Volume Control - Boric Acid Mix System
16160-26018	4 of 8	8	P&ID Chemical & Volume Control - Charging & Metering Pumps
16103-36018	5 of 8	3	P&ID Chemical & Volume Control - Return Line to Reactor Coolant Pump Seals
16103-26018	6 of 8	4	P&ID Chemical & Volume Control - Return & Drain Lines for Reactor Coolant Loops
16103-26018	7 of 8	2	Operations Flow Diagram - Chemical & Volume Control System
16103-26018	8 of 8	3	Operations Flow Diagram - Chemical & Volume Control System

2. Haddam Neck Valve IST Program, Dated January, 1987 (2nd ten-year interval for the IST Program)

3. Haddam Neck System Descriptions for Interfacing Systems

   a) Chapter 4, "Chemical and Volume Control System", Rev. 0, dated 4/30/87

   b) Chapter 5, "Emergency Core Cooling System", Rev. 0, dated 4/1/86

   c) Chapter 6, "Residual Heat Removal System", Rev. 0, dated 5/6/87

4. Stone & Webster Specifications

   a) Specification No. CYS-1550, "Specification for Shop Fabricated Nuclear Piping", revised July 21, 1965

   b) Specification No. CYS-579, "Specification for Shop Fabricated Piping for Secondary Plant and Primary Waste Disposal and Other Miscellaneous Systems", revised December 10, 1965

5. NUREG-0700, "Guidelines for Control Room Design Reviews", 1981.

Appendix A (continued)

6. NUREG-0700, "Comparison and Application of Quantitative Human Reliability Analysis Method for the Risk Method Integration and Evaluation Program (RMIEP)", January 1989.

Appendix B - Procedures

1. Surveillance Procedures

Number	Revision	Title	Date
SUR 5.1-1	14	Hydrostatic Test	10/27/87
SUR 5.1-4	22	Core Cooling Systems Hot Operational Test	05/19/89
SUR 5.2-52	9	Reactor Coolant Pressure Channel Calibration	04/11/89
SUR 5.4-34	7	Performance Testing of Reactor Coolant Post Accident Sampling, Module (Test and/or Training)	04/27/89
SUR 5.7-46	7	Loop Drains Header, P-14	06/28/88
SUR 5.7-65	6	Loops Safety Injection, P-3 Isolation Valves SI-CV-862A, B. C and D Local Leak Rate and Pressure Isolation Testing	02/24/89
SUR 5.7-66	5	Safety Injection Recirculation, P-24	02/17/89
SUR 5.7-91	3	RHR Inboard and Outboard Isolation MOVs	12/04/88
SUR 5.7-111	2	Leak Testing of Core Deluge Check Valves	07/23/87
SUR 5.7-128	1	Loop Safety Injection Stop Valves SI-MOV-861A, 861B, 861C and 861D Pressure Isolation Test	06/06/89
SUR 5.7-135	Original	HPSI and LPSI Discharge Check Value Operability Test	07/20/87

2. Maintenance Procedures

Number	Revision	Title	Date
PMP 9.5-4	11	Limitorque Valve Motor Operator Preventive Maintenance	04/24/89
PMP 9.5-215.1	Original	Limitorque Operator Removal Installation, and Adjustment	04/24/89
MDI-22	3	Use of Procedures	10/28/88
MDI-01	Original	Maintenance Department Organization and Administration	10/31/88
MDI-75	Original	Control of Maintenance Activities	04/20/89
MDI-60	Original	Post-Maintenance Cleaning After Valve Seat Maintenance	10/27/88

Appendix B (Cont'd.)

Number	Revision	Title	Date
MDI-63	2	Qualification of Mechanics and Electricians	11/16/88
PMP 9.5-0	4	Maintenance Department Preventive Maintenance Program	08/07/88
MDI-16	23	Preventive Maintenance	04/06/89
ENG 1.7-55	1	Documentation and Evaluation of Inservice Valves Testing	03/22/89
ACP 1.2-11.3	13	Retests/Functional Verification	04/04/89
EDI 3.19	1	Inservice Inspection/Inservice Testing PDCR Review	03/30/89
ACP 1.0-39	2	Repair, Rework, and Replacement Plan for Class I, II, and III Systems or Components, and Associated Supports (RRR)	02/22/89
MA 1.5-1	2	Work Order Preparation, Work Control and Documentation	05/22/89
MDI-36	3	Relief Valve Test Program	10/27/88

3. Emergency Operating Procedures

Number	Revision	Title	Date
EO	6	Reactor Trip - Safety Injection	07/07/89
ECA 1.2	1	LOCA Outside Containment	06/03/88
ECA 1.2 Ho	1	LOCA Outside Containment (WOG ERG)	09/01/83
ES-1.3	7	Transfer to RHR Recirculation	07/07/89
E.1	5	Loss of Reactor or Secondary Coolant	07/07/89
LER 50-213/89-003-00	0	Containment Valve Misalignment	04/24/89
CY-OP-LO-EOP-S033	3	Small Loss of Coolant Accisent	09/09/87
CY-OP-LOCT-87-4-S87401	0	Plant Operations With Malfunctions	02/25/87
CY-OP-LORT-S004	0	Loss of Coolant Accident	09/13/88

Appendix C - Records

1. Surveillance Records

Procedure Number	Testing Performed	Component(s) Tested	Surveillance Test Dates
SUR 5.1-1	Leak Testing	RH-MOV-780, 781, 803, 804	10/30/84, 04/26/86, 03/05/88
SUR 5.7-46	Leak Testing	DH-MOV-310, DH-V-311	01/16/86, 04/22/86, 07/26/87, 10/01/87, 10/17/87
SUR 5.7-65	Leak Testing	SI-CV-862A	01/13/86, 07/30,87, 10/04/87, 01/25/88
SUR 5.7-65	Leak Testing	SI-CV-862B	01/23/86, 04/07/86, 04/14/86, 04/16/86, 04/18/86, 04/19/86, 04/20/86, 07/30/87, 09/17/87, 01/25/88
SUR 5/7-65	Leak Testing	SI-CV-862C, 862D	01/13/86, 07/30/87, 01/25/88
SUR 5.7-66	Leak Testing	SI-V-863A, 863B, 863C, 863D	01/13/86, 07/28/87, 09/16/87 10/01/87
SUR 5.7-111	Leak Testing	SI-CV-872A, 872B	04/27/86, 03/05/88
SUR 5.7-128	Leak Testing	SI-MOV-861A, 861C, 861D	01/13/86, 03/05/88
SUR 5.7-128	Leak Testing	SI-MOV-861B	01/13/86, 04/07/86, 04/14/86, 04/15/86, 04/18/86, 04/19/86, 04/20/86, 03/05/88, 03/10/88
SUR 5.7-64	Stroke Time, Full Exercise, Valve Position	SI-MOV-861A	10/24/84, 01/11/86, 04/23/86, 07/18,86, 09/29/87, 11/10/87, 01/18/88, 01/28/88, 02/01/88, 03/08/88, 05/02/88
SUR 5.7-64	Stroke Time, Full Exercise, Valve Position	SI-MOV-861B	10/24/84, 01/11/86, 04/23/86, 07/18/86, 09/29/87, 01/18/88, 01/28/88, 02/01/88, 03/06/88, 03/08/88, 05/02/88, 05/04/88
SUR 5.7-64	Stroke Time, Full Exercise, Valve Position	SI-MOV-861C	10/24/84, 01/11/86, 04/09/86, 04/23/86, 07/18/86, 09/29/87, 01/18/88, 01/28/88, 02/01/88, 03/06/88, 05/02/88, 05/05/88

Appendix C (Cont'd.)

Procedure Number	Testing Performed	Component(s) Tested	Surveillance Test Dates
SUR 5.7-64	Stroke Time, Full Exercise, Valve Position	SI-MOV-861D	10/24/84, 01/11/86, 04/23/86, 07/18/86, 09/29/87, 11/03/87, 01/18/88, 01/28/88, 02/01/88, 03/06/88, 03/08/88, 05/02/88, 05/13/88, 05/19/88
SUR 5.7-64	Stroke Time, Full Exercise Valve Position	SI-MOV-871A, 871B	10/24/84, 01/11/86, 04/26/86, 07/18/86, 10/28/87, 02/28/88, 05/02/88
SUR 5.7-91	Stroke Time, Full Exercise, Valve Position	RH-MOV-780, 781, 803, 804	02/11/86, 03/18/86, 07/20/86, 08/14/87, 01/18/88, 02/01/88, 05/20/88

2. Others

°Summary of Human Engineering Discrepancies From Detailed Control Room Design Review

°Computer output from the Production Maintenance Management System (PMMS) for valve maintenance history report.