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

SEP 14 1990

Docket No. 50-382

Mr. Ross P. Barkhurst
Vice President-Nuclear
Entergy Operations, Inc.
Post Office Box B
Killona, Louisiana 70066

Dear Mr. Barkhurst:

SUBJECT: INTERFACING SYSTEM LOCA INSPECTION (50-382/90-200)

We are enclosing the report on the interfacing system loss-of-coolant accident (ISLOCA) inspection performed from July 30 through August 10, 1990, at the Waterford 3 nuclear power plant. The Nuclear Regulatory Commission (NRC) staff from the Office of Nuclear Reactor Regulation, Region IV, and NRC contractors conducted this inspection. We discussed the inspection findings with members of your staff during an exit meeting on August 10, 1990.

The NRC inspection team evaluated specific plant design features, systems, equipment, procedures, operations activities, and human actions that could affect the initiation or progress of an ISLOCA. The team focused its review on the high-pressure safety injection, shutdown cooling, and low-pressure safety injection systems, with a cursory review of the chemical and volume control system.

The team found that the pressure isolation valves within systems interfacing with the reactor coolant system pressure boundary at Waterford 3 were adequately maintained and tested to minimize failures that could initiate an ISLOCA. The team did not identify any significant deficiencies in the man-machine interface that might significantly increase the probability of an operator error initiating an ISLOCA. Nevertheless, the team did find some specific deficiencies in the availability of design calculations, check valve maintenance, and plant equipment labeling. The team noted that you are addressing these issues through programs that are planned or currently being implemented.

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No formal response to this report is required. In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Original signed by

Dennis M. Crutchfield, Director
Division of Reactor Projects III/IV/V
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
NRC Inspection Report 50-382/90-200

cc: See page 3

Distribution: See page 4

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ADSP
DMCrutchfield
09/14/90

cc:

W. Malcolm Stevenson, Esq.
Monroe & Leman
201 St. Charles Avenue, Suite 3300

Mr. E. Blake
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, D.C. 20037

Resident Inspector/Waterford NPS
P.O. Box 822
Killona, Louisiana 70066

President, Parish Council
St. Charles Parish
Hahnville, Louisiana 70057

Mr. Donald C. Hintz
Executive Vice President and
Chief Operating Officer
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286

Chairman
Louisiana Public Service Commission
One American Place, Suite 1630
Baton Rouge, Louisiana 70825-1697

Mr. R. F. Burski, Director
Nuclear Safety
Entergy Operations, Inc.
317 Baronne Street
New Orleans, Louisiana 70112

Mr. L. W. Laughlin, Site Licensing
Support Supervisor
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. William H. Spell, Administrator
Radiation Protection Division
Department of Environmental Quality
P.O. Box 14690
Baton Rouge, Louisiana 70898

Mr. Gerald W. Muench
Vice President, Operations
Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286

William A. Cross
Bethesda Licensing Office
3 Metro Center
Suite 610
Bethesda, Maryland 20814

Mr. Robert B. McGehee
Wise, Cater, Child & Caraway
P.O. Box 651
Jackson, Mississippi 39205

Mr. J. R. McGaha, Jr.
General Manager Plant Operations
Entergy Operations, Inc.
P.O. Box B
Killona, Louisiana 70066

Mr. Jerrold G. Dewease
Sr. Vice President
Planning & Assurance
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

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U.S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Division of Reactor Inspection and Safeguards

NRC Inspection Report: 50-382/90-200 License No.: NFP-38

Docket No.: 50-382

Licensee: Entergy Operations, Inc.
Post Office Box B
Killona, Louisiana 70066

Facility Name: Waterford 3

Inspection Conducted: July 30 through August 10, 1990

Inspection Team: Jay R. Ball, Team Leader, NRR
Sammy S. Diab, System/Risk Analyst, NRR
Philip C. Wagner, Inspector, Region IV

NRC Consultants: J. L. Auflick, Idaho National Engineering Laboratory
C. C. Baker, Carlow Associates, Inc.
P. W. Eshleman, Engineering & Science Associates, Inc.
S. A. Fleger, Carlow Associates, Inc.
D. C. Ford, Resource Technical Services
D. L. Jew, EAS Energy Services
D. L. Kelly, Idaho National Engineering Laboratory
O. R. Meyer, Idaho National Engineering Laboratory

Approved by: Jay R. Ball 9/7/90
Jay R. Ball, Operations Engineer Date
Team Inspection Development Section A
Special Inspection Branch
Division of Reactor Inspection and Safeguards
Office of Nuclear Reactor Regulation

Approved by: Wayne D. Lanning for 9/10/90
James E. Konklin, Chief Date
Team Inspection Development Section A
Special Inspection Branch
Division of Reactor Inspection and Safeguards
Office of Nuclear Reactor Regulation

Approved by: Wayne D. Lanning 9/10/90
Wayne D. Lanning, Chief Date
Special Inspection Branch
Division of Reactor Inspection and Safeguards
Office of Nuclear Reactor Regulation

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EXECUTIVE SUMMARY

The NRC team conducted an interfacing system loss-of-coolant accident (ISLOCA) inspection at the Waterford 3 nuclear power plant from July 30 through August 10, 1990. This inspection supported the ongoing NRC program for assessing the probability for ISLOCAs at operating nuclear power plants. The objective of the inspection was to collect data and information about plant conditions, including design features, systems, equipment, procedures, and operations that could affect operator detection, diagnosis, and response to an ISLOCA. In addition, the team collected information related to human reliability analysis (HRA) for a study being conducted by NRC's Office of Nuclear Regulatory Research.

ISLOCA refers to a type of postulated event in which the pressure boundary between the reactor coolant system (RCS) and a low-pressure system is breached, resulting in a loss of primary coolant outside containment. The team focused on the shutdown cooling and safety injection systems because of the possible consequences of component failures or operator errors in those systems. The team limited its review of the chemical and volume control system because of the low probability of an ISLOCA in that system.

The team found that the pressure isolation valves (PIVs) within systems interfacing with the RCS pressure boundary at Waterford 3 were adequately maintained and tested to minimize failures that could initiate an ISLOCA. The team did not identify any significant deficiencies in the man-machine interface that might significantly increase the probability of an operator error initiating an ISLOCA. However, the team identified weaknesses in the man-machine interface that could adversely affect the ability of the operators to mitigate an ISLOCA because of poor equipment labeling and the inaccessibility of some equipment.

The team identified one scenario involving a normal cooldown evolution that appeared to have a higher than expected probability for occurrence. A simulator exercise demonstrated that the operators, although not specifically trained and lacking specific procedural guidance, were able to adequately cope with the event.

The team considered that the lack of existing design calculations to verify the ability of PIVs to close against postulated differential pressures was a weakness. In addition, no calculation existed that showed check valves located within the suction line from the reactor water storage pool (RWSP) to the low-pressure safety injection (LPSI) pumps were correctly positioned with respect to upstream pipe fittings to ensure that they would not become damaged as a result of flow turbulence. The licensee performed an preliminary calculation during the inspection that showed acceptable positioning of the valves.

The team found the licensee's maintenance program for the PIVs generally effective and considered the failure trending and analysis to be a strength. Although the licensee had developed an adequate check valve maintenance program, the team noted a weakness in the maintenance of two LPSI pump suction header check valves. These valves had recently been included in the program but had not received any form of internal maintenance or inspection during the life of the plant.

The team concluded that the surveillance performed on PIVs was appropriate. Although minor weaknesses were found in the surveillance procedures, these had already been identified by the licensee and the procedures were in the process of revision.

The team found emergency operating procedures to be well written although they lacked some human factors considerations. In addition, annunciator response procedures were found to contain some inconsistencies in format and wording.

Although training specific to ISLOCAs was not a part of the licensee's training program, operators indicated, during walkthroughs and simulator exercises conducted by the inspection team, that they were generally well prepared to cope with losses of RCS inventory.

1.0 BACKGROUND

The interfacing system loss-of-coolant accident (ISLOCA) is a postulated loss-of-coolant accident in which an interface between the high-pressure boundary of the reactor coolant system (RCS) and connecting low-pressure piping is breached. This type of accident is of special concern because overpressurization of the low-pressure systems could result in a rupture outside containment and thereby a discharge of reactor coolant to the environment. Furthermore, mitigation systems for all types of LOCAs could be adversely affected by an ISLOCA.

The ISLOCA was first identified as a significant contributor to risk in the Reactor Safety Study WASH-1400. The ISLOCA was then referred to as Event-V, and was limited to the failure of two check valves (the pressure isolation valves) which lead to overpressuring and rupture of the low pressure system. The ISLOCA has now been expanded to include failure or inadvertent opening of motor-operated valves. The consequences of an ISLOCA are greatly dependent on plant features, break locations, and mitigating actions, and are associated with many uncertainties. Thus, the Nuclear Regulatory Commission (NRC) initiated a series of inspections by the Office of Nuclear Reactor Regulation (NRR) and related efforts by the NRC Office of Nuclear Regulatory Research (RES) to collect information on plant features that could affect the frequency and severity of an ISLOCA. ISLOCA inspections have been performed at several other nuclear power plants. The team used the results of these inspections to prepare for the Waterford 3 ISLOCA inspection.

2.0 INSPECTION APPROACH

2.1 Objective and Scope

The primary objective of this inspection was to evaluate specific plant design features, systems, equipment, procedures, operations activities, and human actions that could affect the initiation or progress of an ISLOCA. This included identification of generic events or system features associated with postulated ISLOCAs, possible initiating or precursor events, and possible related human errors.

The team assessed licensee programs relevant to the ISLOCA and reviewed various licensee records to determine the effectiveness of preventive, corrective, and mitigative measures. The team considered pressure isolation valves (PIVs) to be those that isolated the higher pressure RCS from the lower pressure interfacing systems. The team focused its review on the shutdown cooling (SDC), low-pressure safety injection (LPSI), and high-pressure safety injection (HPSI) systems because of their importance to the ISLOCA scenarios and potential consequences. The team developed ISLOCA scenarios for each of these systems. Multiple failures of equipment, inadequate procedures, and human error were considered in this development process. To a limited extent, the team also reviewed the chemical and volume control system (CVCS) and scenarios related to failures in that system.

The systems considered for the ISLOCA inspection are discussed below, followed by a discussion of the scenarios developed to identify conditions that could affect initiation or progress of an ISLOCA. Sections 3, 4, and 5 address the detailed inspection results, and Section 6 provides the overall conclusion of the inspection team. Appendix A lists the persons attending the exit meeting held with the licensee's representatives on August 10, 1990. Appendices B, C, and D provide supplemental information to this inspection report.

2.2 System Descriptions

The systems considered for this inspection generally met the following criteria:

- piping that was connected to the RCS and penetrates the containment (lines that were connected to the RCS but do not penetrate the containment were not considered because ruptures in these lines would result in a LOCA inside the containment, which was a design-basis accident.)

- interfacing piping with design pressure ratings substantially below the RCS pressures.

- associated piping with the capacity for a sufficiently large leak rate so that the normal makeup system would not have the capability to replace the inventory lost.

The interfacing systems satisfying these criteria at the Waterford 3 plant were the HPSI, the safety injection tank (SIT), the SDC in conjunction with the LPSI and the CVCS.

Low-pressure systems were postulated to be overpressurized by valve manipulating or failures such as valves left open after scheduled surveillance or maintenance, inadvertent valve opening by operators, spurious valve opening, or a combination of these mechanisms.

2.2.1 High-Pressure Safety Injection System

The function of the HPSI system was to inject cooling water into the RCS during small- and medium-size LOCAs through cold- or hot-leg injection flow paths. The cold-leg injection mode is started automatically upon receipt of a safety injection signal. Hot-leg injection is activated by the operator a few hours into the event to avoid boron precipitation.

Two HPSI pumps, A and B, are aligned to inject water into the four cold-leg flow paths. An additional swing pump A/B can be manually aligned to provide for injection. In each of the HPSI flow paths, there were two check valves inside the containment and a normally closed motor-operated valve (MOV) in the reactor auxiliary building (RAB). Injection into the hot-leg flow paths can be aligned manually through two injection lines. Each of the injection lines had two check valves inside the containment and two closed MOVs in the RAB. The MOVs in the hot-leg injection flow path were procedurally controlled and locked closed with key-operated control switches in the control room. Appendix B, Figure 1, shows a diagram of the HPSI system and the relative location of the HPSI components within the RAB and the containment building.

The closed MOVs were the high-to-low pressure interfaces. The containment penetration piping downstream from the closed MOVs, including the two check valves, had a design pressure of 2485 psig. The piping in the RAB upstream of the MOVs up to the pump discharge had a design pressure of 1950 psig. All the normally closed MOVs on the HPSI injection lines that were in the RAB were easily accessible for local operation. However, operation of these valves could be restricted by a break or by the inability of the valve to close against high flows or pressure differentials if they had been opened.

If an ISLOCA break occurred at the suction portion of an HPSI pump and the refueling water storage pool (RWSP) outlet isolation valves (SI-106 A or B) could not be closed, the RWSP water would be lost. This would affect long-term recovery and core cooling, and valves SI-106 A and B were not readily accessible for local operation.

2.2.2 Safety Injection Tank System

The SIT system was a passive system that provided a large volume of water to make up for lost coolant in the reactor core if a large break LOCA occurred. The system was actuated when the RCS pressure dropped below the pressure of the cover gas in the tanks, which was maintained at about 650 psig.

The SIT system consisted of four pressure vessels filled with borated water and pressurized with nitrogen gas. A 12-inch-diameter outlet line for each SIT was connected to one of the cold legs through a safety injection line. Since the SITs were passive components, only check valves and a locked-open MOV isolated them from the cold-leg flow paths.

The SIT system piping that penetrated the containment consisted of the drain line and the fill lines that also were common to the HPSI and LPSI systems. The drain line was a 2-inch line with two normally closed air-operated globe valves, and a manual valve outside the containment which acted as a 1950/550 psig high-to-low pressure interface. The 550 psig was the design pressure on the outside of the outermost valve. There were numerous valves that could be used to isolate a break in either the drain line or fill lines. However, operation of any of these valves could be restricted by a break or by the inability of the valve to close against high flows or pressure differentials.

High pressure alarms and instrumentation were available to alert the operator to an ISLOCA within the SIT system. SIT pressure and level indications were also provided in the control room so that any in-leakage or out-leakage from the tanks could be detected.

2.2.3 Shutdown Cooling and Low-Pressure Safety Injection Systems

The function of the Shutdown Cooling (SDC) system was to remove decay heat during normal plant shutdowns. The SDC system used the LPSI system lines and LPSI pumps to perform this function. A line from the RCS directed inventory to the LPSI pumps suction header. To remove decay heat, the flow was directed through line connections to the SDC heat exchangers and back to the RCS. The function of the LPSI system was to inject cooling water into the RCS during medium and large break LOCAs. Appendix B, Figure 2, shows a diagram of the SDC flow path from the RCS through the LPSI system and back to the RCS and the relative location of the SDC/LPSI components within the auxiliary and containment buildings. The SDC system was placed in operation when the RCS pressure and temperature were less than 392 psig and 350°F. The LPSI system was normally in standby during power operations and operated automatically upon receipt of a safety injection signal. It could also be actuated manually.

Hydraulically operated valves (HOV) SI-405 A and B in the SDC 14-inch-diameter suction lines provided high-to-low pressure interfaces with the RCS. The design pressure was 2485 psig on the high-pressure side (RCS side) and 440 psig on the low-pressure side for components and penetration piping. In addition, MOVs SI-401 A and B with a 2485-psig pressure rating and MOVs SI-407 A and B with 440-psig pressure rating were located in the flow path from the RCS to the LPSI pump suction and were normally in the locked-closed position. These valves served as additional isolation valves for the SDC suction lines.

Valves SI-401 A and B and SI-405 A and B located inside the containment were equipped with position indications that alarmed in the control room to alert an operator if the valves were off their closed-seat and the RCS pressure was greater than 392 psig.

The above valves were equipped with RCS pressure automatic closure interlocks (ACI). If any of these valves was inadvertently left open during startup operations, the ACI would cause them to automatically close when the reactor pressure increased above 700 psia. However, the licensee planned to remove the ACI because of other considerations. Additionally, during the shutdown evolution, an open permissive interlock (OPI) allowed the valves to be opened from the control room only when the RCS pressure was within the SDC system design pressure.

Pressure relief valves SI-406 A and B were located on 6-inch-diameter low-pressure piping downstream of HOVs SI-405 A and B. SI-406 A and B, which discharged to the containment sump, had a rated capacity of about 3100 gallons per minute and a setpoint of 415 psig.

Each SDC/LPSI injection line to the cold legs had two check valves inside the containment and a normally closed, fail as-is MOV in the auxiliary building. Between the two check valves on each of the four injection lines, there was a pressure transducer that indicated the pressure between the two valves in the control room and caused an alarm to sound if the pressure increased to 1000 psig. Between the closed MOV and the outboard check valve there was a test connection that was used for leak testing. As the RCS is pressurized during startup, the pressure indication/alarm in the control room should alert the operator if the first check valve was leaking or had failed open. To isolate an ISLOCA in the SDC/LPSI lines, there were a number of valves that could be operated from the control room or at the valve locations. However, the ability to close these valves could be restricted by high flow rates or pressure differentials.

The SDC/LPSI MOVs and manual valves located outside the containment were readily accessible for local operation, with the exception of SI-407 A and B. MOVs SI-407 A and B were located about 20 feet above the floor and had small handwheels that could be difficult to manipulate. Locked-open manual valves SI-410 A and B were in line, respectively, with SI-407 A and B and could possibly be used to isolate an ISLOCA after being unlocked. If an ISLOCA break occurred at the suction portion of an LPSI pump and valves SI-106 A or B could not be readily closed, the RWSP water would be lost, which would affect long-term recovery and core cooling.

2.2.4 Chemical and Volume Control System

The functions of the CVCS included automatic control of the RCS inventory and control of the boron concentration and reactor water purification.

The two CVCS lines that penetrated the containment were the letdown and charging lines. The letdown line was a 2-inch line with a number of air-operated valves that fail closed on loss of air or power. The second valve outside the containment (the letdown flow control valve) acted as the high-to-low pressure interface. The penetration piping up to and including the letdown flow control valve had a design pressure of 2485 psig, and the piping downstream of that had a design pressure of 650 psig.

The discharge lines from three positive displacement pumps combine into one injection line which penetrated the containment at the location of a locked-open valve, entered the regenerative heat exchanger, then lead into four branch lines. Two of the branch lines, which each has a normally closed solenoid-operated valve and a check valve, combined into one header and fed the pressurizer auxiliary spray. The other two branches, which each has normally open solenoid-operated valve and a check valve, fed two of the cold legs. The piping downstream of the charging pumps had a design pressure of 3125 psig.

2.3 ISLOCA Scenarios

The team reviewed seven ISLOCA scenarios to identify conditions that could affect initiation or progress of an ISLOCA. These scenarios involved a failure of two pressure isolation valves in series, resulting in a loss of pressure boundary and subsequent overpressurization of a low-pressure (i.e., less than RCS pressure) system. Various other failures were then postulated, such as relief valve failures, loss of pump seals, or rupture of various piping or flange connections. The circumstances and assumptions underlying the postulated ISLOCA scenarios involved multiple failures and exceedance of the plant's design bases. The seven scenarios are listed below.

1. Failure of SDC suction isolation valves

2. Failure of LPSI cold-leg discharge check valves

3. Failure of HPSI cold-leg discharge check valves

4. Failure of HPSI hot-leg discharge check valves

5. Failure of charging system cold-leg discharge valves

6. Failure of the letdown flow control valve

7. Failure of check valves in the suction line from the RWSP to the LPSI pumps.

The team considered the seventh scenario to have the greatest chance for occurrence. The seventh scenario involved a normal plant cooldown evolution that would result in challenging two normally unseated check valves located in the line between the RWSP and the suction portion of one of the LPSI pumps.

These check valves (SI-108 A or B and SI-1071 A or B) separated piping rated at 160 psig from piping rated at 440 psig. During a plant cooldown, when transitioning from Mode 4 to 5, the operators place one of the two trains of shutdown cooling into service by opening the three isolation valves (SI-401 A or B, SI-405 A or B and SI-407 A or B) between the RCS hot legs and the suction portion of a LPSI pump. This occurred by procedure when the RCS pressure was no greater than 392 psig. When the last of these valves was opened, the check valves in the suction line from the RWSP must seat to prevent over-pressurization of the upstream piping. If the valves failed to seat, the low-pressure piping could fail as a result of overpressurization. RCS inventory would be lost unless operator action was taken to reclose at least one of the three isolation valves in the SDC suction line. In addition, the scenario indicated that, if the operators were unable to close the RWSP outlet isolation valve (SI-106 A or B), which was a normally open, fail-as-is, air-operated butterfly valve, the contents of the RWSP could drain to the basement of the reactor auxiliary building. This could affect long-term recovery and core cooling.

3.0 PLANT DESIGN FEATURES

3.1 Design Capability of Isolation Valves

The team evaluated the electrical and mechanical design characteristics of various pressure isolation valves (PIVs) to determine their ability to prevent or terminate an ISLOCA. The team's review of the electrical schematic and the control wiring diagrams showed that the control and interlock functions associated with the operation of the valves were satisfactory. The team also found the equipment qualification data records for selected PIVs to be satisfactory.

The team reviewed in detail SDC suction header isolation valves (SI-401 A and B and SI-405 A and B). The motor-operated SI-401 A and B valves isolated their SDC loop from the RCS near the penetration to the respective RCS hot leg. The pneumatic, hydraulically operated valves SI-405 A and B performed a fast-acting isolation function and were located immediately downstream of the respective SI-401 valve.

The team evaluated the capabilities of the SDC valve actuators to close the valves against their design pressure and flow rate conditions. The valve data sheet for the SI-401 A and B valves indicated that the stem thrust necessary to close the valves at their design pressure of 2485 psig had been calculated to be 64,016 pounds. The team performed an independent calculation using more current industry data and determined that the required stem thrust could be as high as 115,000 pounds. The team discussed these calculations with licensee personnel and were told that the SI-401 A and B valves were included in the motor-operated valve testing program and that they would be reevaluated as part of that program. In addition, licensee personnel stated that these valves would not be open when the RCS was pressurized above 700 psig. The team verified that open permissive interlocks and automatic closure features were incorporated into the control circuitry for the valves. The team also reviewed the administrative controls that limited the open condition of the valves. These reviews are discussed in more detail in Section 3.2.

The SI-405 A and B valves were opened by hydraulic oil pressure acting below the actuating piston and were closed by gas pressure acting above the piston. The team questioned the ability of the pneumatic-hydraulic actuators to close the valves if the volume of nitrogen gas in the accumulators was at the low-pressure alarm setpoint. The licensee was unable to produce any existing calculations that proved the ability of the valves to close. Therefore, licensee personnel performed calculations of the stem thrust required to close the SI-405 A and B valves at various RCS pressures. The licensee's calculations indicated that a stem thrust of 21,478 pounds would be required to close the valves with an RCS pressure of 700 psig. The licensee then calculated the gas pressure needed to produce the necessary stem thrust. On the basis of those calculations, the licensee determined that a significant margin existed even if valve closing was initiated at the low-pressure alarm setpoint. Independently the team performed a calculation and determined that a stem thrust of approximately 40,000 pounds would be required to close the valves if the RCS pressure were 700 psig. This calculation was based on the latest NRC guidance, which may not have been used in the licensee's calculation. However, the team verified that adequate pressure would be available to close the valves in question.

The team concluded that the SDC isolation valves would function properly under all of the postulated operating conditions. However, the lack of existing design calculations to verify the ability of PIVs to close against postulated differential pressures was considered a weakness. An additional case of missing calculations is discussed in Section 4.2.1.

3.2 Shutdown Cooling Suction Valve Interlocks

The team evaluated the technical adequacy of the shutdown cooling (SDC) system safety-related interlock circuits by reviewing a limited number of loop calculations, design changes, elementary wiring diagrams, schematics, modification packages, and equipment specifications. The team conducted specific reviews of the power and control circuit interconnecting wiring diagrams to verify the independence of sensors, interlock circuits, and power supplies. The team also reviewed the operation, testing, and calibration procedures associated with the SDC system interlocks and the PIVs interconnected to the RCS.

The high-to-low pressure interconnection of the RCS to SDC system was accomplished by redundant trains of two isolation valves. Each valve was controlled by an interlock circuit with an independent pressure sensor from an independent tap on the pressurizer. Vital instrumentation and dc power provided independent control of and power to each of the valves. MOVs SI-401 A and B at the interface to the RCS were normally closed with power disconnected to avert inadvertent operation. A permissive setpoint control existed that allowed valve opening only when RCS pressure was below 392 psig and operator action was required to open the valves. The second valve in each flow path (HOV SI-405 A or B) maintained in an open position only when a low setpoint in pressurizer pressure was present. The valve closed by the interlock circuit action when pressurizer pressure exceeded 700 psig. The interlock operation for these isolation valves was provided from diverse power sources and valve operation was provided from independent power sources, with power failure causing the HOV to close. During normal power operation, all of the isolation valves had either their power disabled or their control switch in a locked-close position.

The interlock circuitry provided protection from the inadvertent opening of the SDC system suction valves when RCS pressure was higher than the SDC system design pressure. In addition, the interlocks provided for the automatic closure of the MOVs and HOVs when the system pressure increased above a high setpoint. The interlocks performed their function whether the plant control was from the control room or the remote shutdown panel. The interlock circuitry was overridden during long periods of SDC operation by disabling the MOVs automatic closure circuitry and mechanically blocking the HOVs in the open position. Administrative controls which documented performance and verification steps were used to ensure the return of the valves to their automatic control status.

During review of SDC system interlocks and related components, the team noted that the P&IDs contained three different component identifiers. Other plant documentation such as operating procedures had component identifiers that differed from those of the Final Safety Analysis Report (FSAR) and the elementary wiring diagrams. The team considered this a weakness because it provided awkward and confusing operational information.

3.3 ISLOCA Annunciator Availability

The team reviewed Operating Instruction OI-002-000, Revision 8, to determine plant annunciator status control. This procedure allowed the shift superintendent to remove alarm windows from service if they were in a continued alarm status. Plant engineering provided a status log weekly to track the addition of disabled and the return of alarm windows to active service. Three windows associated with the selected ISLOCA scenarios were in a disabled condition. Two of these were nuisance alarms that indicated the correct removal of power from isolation valves SI-401 A and B. The third alarm, N0607 "Hot Leg Injection Line Check Valve Leakage," was disabled preventing operator knowledge regarding possible valve SI-301 leakage, which could be a precursor to an ISLOCA event. No compensatory actions had been taken for this window. However, pressure indication, which could alert the operators to excessive check valve leakage, was available in the control room.

The team considered the procedures used to obtain a "blackboard" condition to be a strength in that they reduce the number of alarms that do not provide the operator with meaningful information. However, the team considered the removal of an alarm indicator without establishing a compensating alarm or special watch condition for the disabled alarm to be a procedural weakness.

4.0 MAINTENANCE, SURVEILLANCE, AND TESTING

4.1 Surveillance and Testing

The team identified pressure isolation valves (PIVs) that could affect the initiation or progress of an ISLOCA and reviewed surveillance testing procedures and test results. In addition, the team reviewed the licensee's Technical Specifications, response to Generic Letter (GL) 87-06, and flow diagrams for the RCS, the HPSI system, the SDC/LPSI systems, and CVCS to ensure that the licensee had identified two valves in each system as PIVs at all appropriate high- to low-pressure interfaces.

4.1.1 Pressure Isolation Valve Identification and Classification

Technical Specification Table 3.4-1, "Reactor Coolant System Pressure Isolation Valves," the licensee's response to GL 87-06 dated June 11, 1987 and Appendix C to this inspection report provide a list of PIV's along with additional information on testing.

Small 1-inch globe valves that served as isolation valves for bypass lines around the PIVs for the HPSI discharge header, hot-leg injection flow path, and LPSI header B discharge were not considered to be PIVs by the licensee. The bypass lines permitted individual leak rate determination for each PIV. Because of the small size of the isolation valves, the team did not consider the licensee's failure to leak test these valves as PIVs to be a concern.

Six additional drain valves for the SITs were located in the system such that they also could be construed to be PIVs. Again, four of these six valves were small 1-inch globe valves, therefore, the team did not consider them to be a concern with regard to excessive leakage. The remaining two valves, SI-301 and SI-302, were 2-inch globe valves. Although leakage through these valves could be in excess of normal RCS makeup capability should gross failure occur, these two valves were located inside containment with the downstream piping that would become overpressurized should valve failure occur also inside containment. Therefore, the team did not consider the failure of these valves and associated piping to be a concern because such failure would result in a LOCA inside containment, which is a design-basis accident.

The team concluded that the licensee's identification and classification of PIVs was satisfactory.

4.1.2 PIV Surveillance Testing

The leak testing of all licensee-identified PIVs was performed using Surveillance Procedure OP-903-008, Revision 2, "Reactor Coolant System Isolation Leakage Test." This procedure was well written, concise, and technically adequate to determine accurate leakage rates for most of the PIVs. It provided proper valve lineup for the tests, use of appropriate equipment and instrumentation to determine leakage, and the establishment of proper test conditions and acceptance values. The team, however, identified the following weaknesses within the procedure:

° Leakage for SIT discharge valves SI-329A, -329B, -330A, and -330B was determined by the change in level of the appropriate SIT over a specific time frame. SIT level indication was provided by computer display in the control room. However, inaccuracies of loop instrumentation could result in a variance of the final leakage results by as much as 0.25 gpm or 25 percent of the acceptance criteria of 1 gpm. The team reviewed the surveillance test results for these four valves and found that no leakage approached 0.75 gpm.

° Leakage past the four cold-leg injection PIVs (SI-335A, 335B, 336A, and 336B) and two hot-leg injection PIVs (SI-512A and 512B) was determined by measuring the change in pressure upstream of the valves (i.e., on the low-pressure side) over a specified time frame and converting this change in pressure into gpm leakage. This method appeared not totally adequate because it was predicated on the assumption that PIVs and smaller drain valves outside the injection lines were leaktight. Should any leakage existed for these other valves, the leakage test results for the PIVs would be nonconservative.

The team discussed these weaknesses with the licensee and was told that the licensee had previously identified similar concerns and had initiated a revision to the applicable surveillance procedure to correct these weaknesses and incorporate other minor changes.

The team concluded surveillance tests had been performed adequately at the required time intervals. Furthermore, any valves that had undergone significant maintenance had had an acceptable post-maintenance leak test performed.

4.1.3 Relief Valve Testing

SDC relief valves SI-406 A and B, which were associated with the ISLOCA scenarios, undergo testing in accordance with ASME Code Section XI, Article IWV-3510. Other HPSI, LPSI, and CVCS safety relief valves associated with ISLOCA scenarios also were bench tested to verify set pressure and seat tightness per Maintenance Procedure MM-007-001, "Safety and Relief Valve Bench Testing."

4.2 Maintenance

To determine the effectiveness of the licensee's implementation of maintenance activities associated with PIVs that could affect the initiation or progress of an ISLOCA, the team reviewed maintenance histories, associated maintenance procedures, completed work packages, industry standards, and vendor manuals.

4.2.1 Corrective Maintenance

Maintenance histories indicated that several pressure isolation and check valves had experienced problems such as leakage at valve packing glands or at flanges. Additionally, concerns had been identified by the licensee with regard to remote operation of leakage drain valves and other minor component deficiencies. Selected plant work authorization (WA) packages and associated data sheets indicated that corrective maintenance activities had been conducted in accordance with requirements and had been generally effective in resolving valve leakage and associated component deficiencies.

However, valve maintenance histories indicated that LPSI pump suction header check valves SI-108 A and B had not received any form of internal maintenance or inspection during the life of the plant. Discussions with maintenance personnel disclosed that these valves had not been included in the licensee's program for check valve maintenance until 1989. Interviews with operations personnel further revealed that the licensee had experienced external leakage problems at these valves for over 2 years. Plant operators also expressed concern with regard to the reliability of these check valves. During the walkdown of system components, the team noted that the area immediately surrounding these valves had been designated a high-radiation area and access to the valves was controlled accordingly. Additionally, repeated attempts to eliminate valve flange leakage had been unsuccessful. The team was concerned that the lack of preventive and corrective maintenance and the material condition of these valves could affect the initiation or mitigation of an ISLOCA in the LPSI pump suction line. These conditions were considered to be a weakness.

The SI-108 A and B valves were dual-plate wafer check valves manufactured by TRW Mission Inc. The vendor technical manual indicated that spacing of this valve type from upstream pipe fittings was critical to prevent damage as a result of turbulence under certain flow conditions. Plant isometric drawing E-2803-IC-63 and team walkdowns of the associated lines indicated that the required spacing may not have been obtained for these valves. In response to the team's concern, the licensee performed an informal calculation that indicated the valves would experience a flow velocity less than the value determined by the vendor to adversely affect the integrity of the valves. The team considered the absence of an existing calculation showing this to be the case to be a weakness in the area of design engineering.

Valves SI-1071 A and B were also components that possibly could affect one of the ISLOCA scenarios investigated. However, the plant engineering personnel could not readily locate documentation for the reason these valves were added to the design or for their design limitations and qualification status, including recent test results. The team considered this to also be indicative of a potential weakness in the area of engineering and technical support.

4.2.2 Preventive Maintenance

The team's review of the plant lubrication, MOV diagnostic signature analysis, check valve maintenance, and post-maintenance testing programs indicated that maintenance planning and work activities had been appropriately implemented. Procedures detailing valve maintenance sufficiently incorporated the requirements of vendor technical manuals, industry standards and regulatory guidance. Administrative procedure MD-001-029, Revision 1, "Check Valve Monitoring, Maintenance and Trending Program," provided a mechanism for monitoring and detecting degradation of check valves before possible failure. Check valves were monitored on a frequency not to exceed once every three fuel cycles. Additionally, the procedure required an accelerated frequency of valve inspection should signs of degradation appear. While this procedure was only recently implemented, it appeared to provide an adequate basis for evaluation of vital system check valves.

The team determined that preventive and corrective maintenance procedures provided sufficient technical detail and clarity to perform maintenance activities on PIVs. The format and content of procedures were consistent and generally conformed to the requirements of the maintenance procedure writers guide. The team did not identify any significant deficiencies in the licensee's preventive maintenance program. Furthermore, the team observed that the licensee had significantly decreased the backlog of maintenance activities during the past 2 years.

4.2.3 Plant Material Condition

The team conducted several tours of the plant during the inspection to observe and assess the material condition of the plant. The team's plant walkdowns focused on portions of the HPSI and SDC/LPSI systems. The material condition of the plant generally was adequate, although there were a large number of valves with catch basins throughout the plant. Good housekeeping was in evidence and the areas examined were free of obstructions. Several of the selected motor-operated and check valves examined exhibited minor leaking and boric acid buildup at packing gland, stems, or valve flanges. Procedure OP-100-002, Revision 4, "Leakage Reduction," required that radioactive leaks be identified on a condition identification (CI) report and a CI tag be attached to the affected equipment. In all but three instances, the required CI tags had been attached to the valve and appropriate catch basins had been mounted to restrict and direct the flow of radioactive materials. However, valves SI-503B, SI-506B, SI-226B and HPSI pump A exhibited leakage but did not contain the required CI tags. In response to this observation, the licensee issued CI reports and associated tags to track component leakage.

4.2.4 Failure Trending and Root-Cause Analysis

Administrative procedures UNT-006-003, Revision 0, "Equipment Failure Trending," and UNT-007-025, Revision 2, "Plant Trending Program," provided guidance for tracking and following up adverse trends in personnel, plant, and component performance. The licensee had taken an aggressive, formalized approach to trending component and activity failures. The licensee trended the performance of the various plant systems and equipment, as well as the performance of plant operators and technicians. Equipment performance trending was done by the department responsible for the system, equipment, or component in question. Each department prepared adverse trend reports and the appropriate followup actions. The licensee collected all the trend reports for each quarter in one quarterly report. These reports indicated that problems with several of the valves included in ISLOCA scenarios that had been identified by the licensee.

The licensee had implemented a long-term reliability program (outlined in Plant Directive 40) that provided for trending of significant recurring problems and established a committee chaired by the Assistant Plant Manager for Operations to prioritize significant issues and recommend followup action to the Plant Manager. This program appeared to be an effective tool to keep management aware of significant recurring problems that could affect safety, reliability, and performance of plant components and systems.

The licensee had formalized methodology for collecting and addressing operational experience. An "Events Analysis, Reporting and Response" group was charged with carrying out events analyses and reporting, root cause identification, failure trending, and reliability and availability engineering. All significant operating occurrences in the plant were screened to identify appropriate root causes. Identification of deficiencies was the responsibility of all nuclear operations personnel. Deficiency identification was documented through a number of mechanisms, including significant occurrence reports, nonconformance condition identification (e.g., defective equipment), potentially reportable events (possible licensee event reports) and quality notices (e.g., procedural noncompliance or deficiency).

The characteristics of the root-cause identification process included a root cause determination, root-cause investigation, and corrective action confirmatory review and oversight. If a root cause was determined to have occurred previously (i.e., recurring problem) the issue was reviewed by the management and a significant quality notice was prepared.

The licensee implemented a human performance trending program, which was intended as a management tool to identify a decline in the performance of any department, in accordance with procedure UNT-006-018, Revision 0, "Human Performance Trending." The procedure provided a caution not to directly compare the performance of one group with that of other groups, but to compare a group's current performance with its own previous performance.

The team concluded that the licensee's use of a formalized approach to problem identification, trending, and root-cause analysis was conducive to reliable plant operation. This effort by the licensee was considered a strength and should heighten plant personnel awareness of system reliability and human performance.

5.0 HUMAN FACTORS AND HUMAN RELIABILITY

5.1 Human Factors

The team reviewed man-machine interface, procedures and documents, and operator training to identify instances in which a human error could affect initiation, detection, or mitigation of an ISLOCA event.

5.1.1 Man-Machine Interface

The man-machine interface (MMI) appeared adequate with regard to minimizing the probability of an operator error initiating an ISLOCA. The control room was quiet and exhibited well-controlled access. The overall impression of the control room was one of professionalism and stability. The MMI for remote shutdown panels appeared adequate.

The design and layout of engineered safeguards panels, which would be used extensively for mitigation of an ISLOCA, would make it difficult for plant personnel to perform the operational tasks required to mitigate an ISLOCA. In particular, meter and valve position indicator lights had glare and vertical boards exhibited some mirror imaging and inconsistent display layout.

Glare on safeguards panel components made it difficult to read vertically oriented meters and panel-mounted handswitches (e.g., hand controllers for SDC valves SI-401 A and B, SI-405 A and B, and SI-407 A and B, LPSI flow meters SI-IFI-0390-A and B, and HPSI flow meter SI-IFI-0311 1A). High readings on meters (pointers at top of scale) and switch labels (valve, motor, fan, etc) were difficult to read.

Mirror imaging on the safeguards panel increased the probability of display substitution errors and visual search time. Groups of displays on the vertical board were mirror imaged although the displays are similarly arranged within these groups. The controls on the off-vertical portion of the control boards were not mirror imaged.

In addition, there was no dedicated recorder provided to monitor the volume control tank (VCT) level, which could be used to detect a loss of RCS inventory. However, the operators did indicate that VCT level was usually trended using the plant computer.

During walkdowns of portions of the reactor auxiliary building (RAB), the team noted that isolation valves SI-106 A and B, located in the safeguards room on elevation -35, were inaccessible from the floor. An operator would require a ladder or scaffold to manually close the valves and there were no ladders available on the -35 level of the RAB. Additionally, the 20-foot ladders stored on the -4 elevation could not be used because their large size precluded access to the -35 elevation. In addition, an operator would have to climb a ladder to verify the identification of valves SI-106 A and B because the labeling could not be read from the floor.

Isolation valves SI-407 A and B on the SDC suction line also are mounted about 20 feet above the floor and component identification could not be verified from the floor. If manual closure of the valve was required, an operator would have great difficulty getting to the valves, and operating the valve handwheel and clutch, which would require two hands.

The inaccessibility of these valves, paired with the lack of ladders and component label readability problems, presented substantial obstacles to local operation of the valves. Other plant labeling and identification weaknesses included temporary labeling of vital equipment with marking pen, inconsistent labeling of components and associated references in plant procedures, and control board instrumentation that required the operator to open a spring-loaded label plate to access component identification numbers.

The team concluded that several human engineering designs existed that could adversely affect the ability of the operators to mitigate an ISLOCA. The most notable of which is the inaccessibility and poor labeling of valves in the RAB.

5.1.2 Procedures and Documents

Emergency operating procedures appeared to be well written and in compliance with CEN-152, the Combustion Engineering (CE) Owners Group generic guidance for CE plants. Although the procedures appeared to adhere to the writers guides (i.e., OP-100-013, "Writers Guide for Operating Procedures," and WG-001, "Writers Guide for EOPs"), they were found to be lacking in several human factors areas (e.g., lack of multiple column format, no numbered table of contents, and failure to comply with a standardized plant nomenclature).

The format and wording of the annunciator response procedures (ARPs) for the annunciators in cabinet N differed from the format and wording used for all the other annunciator procedures. For example, "Possible Cause" in one ARP read "SI-401A or SI-405A open before pressure falls below 386 PSIA" while the other ARP (for the annunciator in the other train) read "Isolation valve open and RCS pressure is 386 PSIA." In addition, the ARP section for train A was entitled "Possible effects and Control Room Indicators" while the other ARP for train B had sections entitled "Plant effects/operator actions" and "Indication/Verification." The ARPs referred to control room instruments by citing the indicator number rather than the indicator label. This practice could increase the time it took for an operator to respond to annunciators because the indicator numbers were hidden behind spring-loaded label plates on the main control boards.

In addition, the table of contents for procedures did not provide page numbers to access sections requiring readers to access information by paragraph numbers only.

SDC PIVs SI-401 A and B were closed during plant transition from Mode 5 to 4. By procedure, these valves were required to be closed and breakers SI-EBKR-311A and B in cabinets 8D and 8H were required to be opened and locked into position. A control room operator directed an auxiliary operator to accomplish this activity. The auxiliary operator then confirmed completion of this task through communication with the control room. Locked valve/breaker sign-off sheets were not used when operators repositioned these breakers, and no independent verification occurred which would ensure that the breakers were locked open. This was considered a weakness in the control of locked valves and breakers.

The team concluded that no procedural problems existed that could directly affect the initiation of an ISLOCA under normal operating conditions although weaknesses existed in ARPs and procedures as stated above.

5.1.3 Training

Operators stated that, while they did not recall any training exercises that specifically addressed ISLOCAs, they felt they had been well-prepared to detect and identify breaches of the RCS pressure boundary. Interviews with operators confirmed their ability to describe the symptoms of an ISLOCA and how those symptoms would be indicated in the control room. The licensee had initiated an ISLOCA screening study to identify potential flow paths through which the RCS pressure boundary interfacing with a supporting system of lower design pressure could be breached. Subsequent to this study, a training module was assembled that was designed to increase operator awareness of symptoms of an ISLOCA. Discussions with the licensee indicated that all licensed operators would receive training using this module at least once.

In addition, a simulation scenario capability existed for small LPSI pump leaks in the LPSI pump room. However, according to the simulator supervisor, this scenario had not been implemented as part of operator training. The licensee was able to use the existing simulation scenario to simulate a scenario that very closely paralleled the seventh ISLOCA scenario, which involved failure of the check valves located in the line between the RWSP and the suction portion of one of the LPSI pumps. The team observed an operating crew respond to the scenario on the simulator. Although the crew had presumably never seen this type of event and had little or no procedural guidance on how to handle the specifics of the event, the crew took appropriate actions available to them to mitigate the consequences of the event. However, the lack of any specific procedural guidance or training with regard to this particular scenario did appear to affect the operators' timely coping with the event. It appeared that use of this simulation could enhance training in the area of ISLOCA.

5.2 Human Reliability

The team collected plant-specific data to be used in the NRC's ISLOCA research project. The plant-specific data from Waterford 3 also will be used in the formal ISLOCA probabilistic risk assessment (PRA) and human reliability analysis (HRA) for the plant. HRA models the types of human actions that can either initiate, detect, diagnose, or mitigate potential ISLOCA scenarios. The team's human reliability evaluation focused on collecting detailed information on operator performance as well as plant-specific factors that could increase or decrease the likelihood of operator error (usually called performance shaping factors, PSFs). Typically, these human error probabilities are placed on event trees, which are then used in conjunction with hardware component failure rates from the PRA to determine plant-specific and sequence-specific core melt probabilities. Ultimately, the human reliability analysis becomes an integrated component of the probabilistic risk assessment.

The team reviewed a series of generic ISLOCA-related events as well as plant systems (RCS, LPSI, HPSI, SDCS, CVCS) that could be involved in an ISLOCA. On the basis of this information, the HRA team members collected detailed, plant-specific information using the following methods:

- table-top task analyses of ISLOCA scenarios that were based on structured interviews with operations personnel,

- simulations of several ISLOCA scenarios with detailed observations of crew activities,

- simulator walkthroughs of systems and their corresponding alarms, annunciators, etc. that may be involved in ISLOCA events,

- plant walkdowns of systems in conjunction with licensed reactor operators (shift supervisors, control room supervisors, and nuclear plant operators) and non-licensed nuclear auxiliary operators, and

- detailed review of emergency and abnormal procedures, training lessons, station directives, operating procedures, and performance and surveillance test procedures.

As part of the research project to employ PRA and HRA methods to assess ISLOCA risk, the HRA team members collected plant-specific information relating to PSFs, such as stress, the nature of the task, procedures, training, experience of the operators, and the quality of the man-machine interface. These PSFs can be positive or negative and are used during the detailed quantification of human actions to modify the nominal human error probability assigned to any given human action in a scenario. PSFs also include any recovery factors, such as communications, teamwork, independent verification, and/or system feedback, that would alert operators to critical errors, thereby returning their actions to a "success" (safe) path. Plant data also was acquired to permit assessment of the influence of maintenance and repair on the probabilistic risk of an

ISLOCA. These data included procedures for configuration control, equipment out of service, and mode change checklists. Team observations relating to the quality of work control will also be included.

The ISLOCA PRA, performed by Idaho National Engineering Laboratory (INEL) in conjunction with this inspection, will use the data collected from this inspection to perform an independent analysis of ISLOCA scenarios. Specifically, the analysis will use the RELAP computer code to model system thermal-hydraulic response to overpressurization and will calculate failure distributions for various system piping and components on the basis of the applied stresses induced by the thermal and hydraulic forces. Offsite consequences will be calculated with the MACCS computer code. Plant-specific data include the system piping and instrumentation drawings, piping isometrics, and vendor data on valves, pumps, orifices, heat exchangers, etc.

All of the information gathered by the team will be reviewed by INEL during the PRA process to help ensure that realistic plant-specific assessments are achieved.

6.0 CONCLUSION

The team concluded that the pressure isolation valves within systems interfacing with the RCS pressure boundary at Waterford 3 were adequately maintained and tested to prevent failures that could initiate an ISLOCA. Although there were weaknesses in the man-machine interface, the team did not identify any significant deficiencies that might significantly increase the probability of an operator error initiating an ISLOCA. No unresolved items were identified.

APPENDIX A

Personnel in Attendance at Exit Meeting

Personnel	Organization
Jack Auflick	NRC Consultant, INEL
R.G. Azzarello	Entergy Operations, Director, Engineering & Construction
Dwight E. Baker	Entergy Operations, Director, Operations Support
Jay R. Ball	NRC Team Leader, Special Inspection Branch, NRR
Ronald G. Bennett	Entergy Operations, QA Inspections Supervisor
Timothy P. Brennan	Entergy Operations, Manager, Design Engineering
Steven D. Butler	NRC Resident Inspector-Waterford 3
K.M. Campe	NRC Acting Branch Chief, Risk Appl. Branch, NRR
Albert Cilluffa	Entergy Operations, Maintenance Engineer
Sammy Diab	NRC Team Member, PRAB, NRR
Huu D. Dinh	Entergy Operations, Event Analysis & Reporting
Neil Dubry	Entergy Operations, Senior Engineer
Paul W. Eshleman	NRC Consultant, Engineering & Science Assoc., Inc.
Steven E. Farkas	Entergy Operations, Licensing Engineer
Stephen A. Fleger	NRC Consultant, Carlow Associates, Inc.
Daniel C. Ford	NRC Consultant, Research Technical Service
James G. Hoffpauir	Entergy Operations, Planning & Scheduling Manager
Terry Holman	Entergy Operations, Supervisor, Safety & Engr. Analysis
J.P. Jaudon	NRC Deputy Director, DRS, Region IV
Dennis L. Jew	NRC Consultant, EAS Energy Services
Dana Kelly	NRC Consultant, INEL
J.E. Konklin	NRC Section Chief, Special Inspection Branch, NRR
W.D. Lanning	NRC Branch Chief, Special Inspection Branch, NRR
Larry W. Laughlin	Entergy Operations, Site Licensing Supervisor
Theodore Leonard	Entergy Operations, Acting Manager of Operations & Maintenance
Orville Meyer	NRC Consultant, INEL
P.V. Prasankumar	Entergy Operations, Manager, Technical Services
William T. Russell	NRC Associate Director, NRR
Douglas Schultz	Entergy Operations, Asst. Operations Superintendent
Ward F. Smith	NRC Senior Resident Inspector-Waterford 3
Wayne L. Smith	Entergy Operations, Simulator Supervisor
Philip C. Wagner	NRC Team Member, Region IV

APPENDIX B

Diagrams of the HPSI and SDC/LPSI Systems

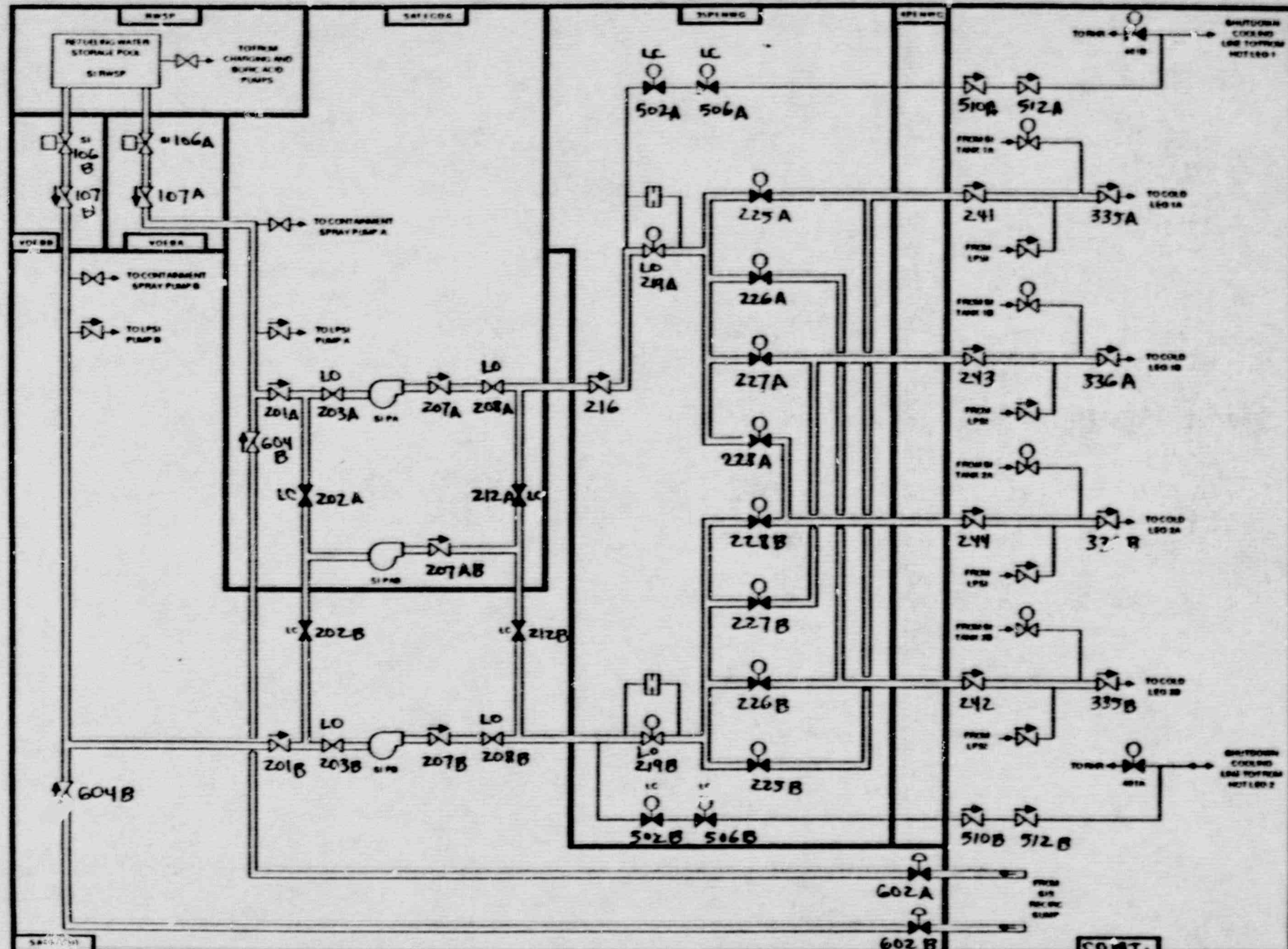

Figure 1. Waterford 3 High Pressure Safety Injection System Showing Component Locations

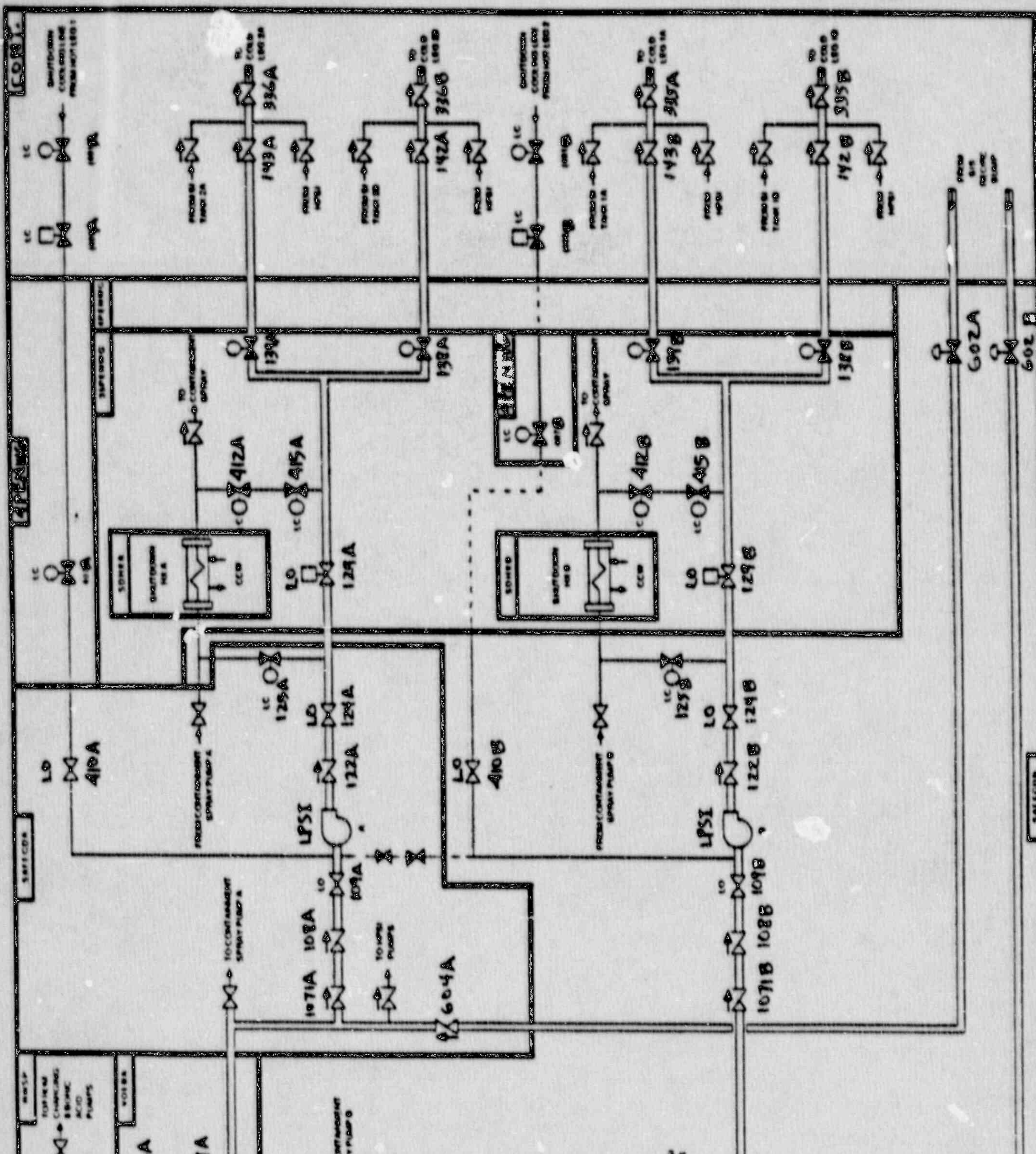

Figure 2. Waterford 3 Low Pressure Safety Injection System Showing Component Locations

APPENDIX C

Pressure Isolation Valves

Valve No.	Description	Type	IST CAT	Required Testing	Alternate Testing	Notes
SECTION A						
SI-329A	SIT Discharge	Check	A/C	CV/PIV	DRR	1
SI-329B	SIT Discharge	Check	A/C	CV/PIV	DRR	1
SI-330A	SIT Discharge	Check	A/C	CV/PIV	DRR	1
SI-330B	SIT Discharge	Check	A/C	CV/PIV	DRR	1
SI-336A	Cold-Leg Inj.	Check	A/C	CV/PIV	DRR	2,3
SI-336B	Cold-Leg Inj.	Check	A/C	CV/PIV	DRR	2,3
SI-335A	Cold-Leg Inj.	Check	A/C	CV/PIV	DRR	2,3
SI-335B	Cold-Leg Inj.	Check	A/C	CV/PIV	DRR	2,3
SI-510A	Hot-Leg Inj.	Check	A/C	CV/PIV	RR	4
SI-512A	Hot-Leg Inj.	Check	A/C	CV/PIV	RR	5
SI-510B	Hot-Leg Inj.	Check	A/C	CV/PIV	RR	4
SI-512B	Hot-Leg Inj.	Check	A/C	CV/PIV	RR	5
SI-241	HPSI Header Disch.	Check	A/C	CV/PIV	RR	4
SI-242	HPSI Header Disch.	Check	A/C	CV/PIV	RR	4
SI-243	HPSI Header Disch.	Check	A/C	CV/PIV	RR	4
SI-244	HPSI Header Disch.	Check	A/C	CV/PIV	RR	4
SECTION B						
SI-142A	LPSI Header Disch.	Check	A/C	CV/PIV	CSR	3,6
SI-142B	LPSI Header Disch.	Check	A/C	CV/PIV	CSR	3,6
SI-143A	LPSI Header Disch.	Check	A/C	CV/PIV	CSR	3,6
SI-143B	LPSI Header Disch.	Check	A/C	CV/PIV	CSR	3,6
SECTION C - POWER-OPERATED VALVES						
SI-401A	SDC Suct. Isol.	Gate	A	Q/MT/PIV	CS	3,7,8
SI-401B	SDC Suct. Isol.	Gate	A	Q/MT/PIV	CS	3,7,8
SI-405A	SDC Suct. Isol.	Gate	A	Q/MT/PIV	CS	3,7,8
SI-405B	SDC Suct. Isol.	Gate	A	Q/MT/PIV	CS	3,7,8

Testing Parameters:

CV Exercise check valve to the position required to fulfill its function at least once every 3 months.

PIV RCS PIVs are leak tested per plant Technical Specifications.

Q Exercise valves (full stroke) for operability at least once every 3 months except when one train of a redundant system is inoperable. Valves in the remaining train should not be cycled because their failure would cause a loss of total system function.

MT Stroke time measurements are taken and compared to the stroke time limiting value per ASME Code Section XI, Article IWV-3410. Trending of valve stroke time is performed per IWV-3417 for valves with stroke time limits greater than 2 seconds.

DRR Valves are disassembled and stroked during reactor fueling outages on a sampling basis.

RR Exercise valve for operability at each reactor refueling outage.

CSR Exercise check valve (partial stroke) at each cold shutdown and full stroke at each reactor refueling outage.

CS Exercise valve (full stroke) for operability during cold shutdown and at each refueling outage.

Notes:

1. One of three valves (SI-329A, -329B, and -330) and valve SI-330A will be disassembled and manually exercised every refueling outage per Relief Request 3.1.16.

2. One of four valves (SI-335A, -335B, -336A, and -336B) will be disassembled and manually exercised every refueling outage per Relief Request 3.1.18.

3. When corrective action is required, a retest will be satisfactorily performed before the valve is required for plant operability as defined in the plane Technical Specifications per Relief Request 3.1.3.

4. These valves will be full-stroke tested during each refueling outage per Relief Request 3.1.14.

5. These valves will be full-stroke tested during each refueling outage per Relief Request 3.1.20.

6. These valves will be partial-stroked tested during each cold shutdown and full-stroked using LPSI design flow during each refueling outage per Relief Request 3.1.13.

7. If increased stroke time exceeds the criteria of IWV-3417(a), the test frequency shall be increased to once each cold shutdown, not to exceed once each month per Relief Request 3.1.4.

8. These valves shall be full-stroked tested for operability at each cold shutdown per Relief Request 3.1.19.

APPENDIX D

Documents Reviewed

1. Administrative Procedures

Procedure No.	Title	Rev.	Date
UNT-005-002	Condition Identification	9	9/7/89
UNT-005-004	Temporary Alteration Control	7	12/12/89
UNT-005-007	Plant Lubrication Program	4	7/10/90
UNT-005-010	Independent Verification Program	1	N/A
UNT-005-011	Calibration and Control of Measuring and Test Equipment	2	3/27/89
UNT-005-015	Work Authorization Preparation and Implementation	1	9/5/89
UNT-006-003	Equipment Failure Trending	0	1/16/90
UNT-007-025	Plant Trending Program	2	2/19/90

2. Operating/Surveillance Procedures

Procedure No.	Title	Rev.	Date
OI-002-000	Annunciator and Alarm Status Control	6	9/8/89
OI-006-000	Operator Aids, Use & Control	3	12/8/78
OP-001-001	RCS Fill and Vent	9	6/1/90
OP-001-003	RCS Drain Down	10	7/30/90
OP-002-005	Chemical & Volume Control	9	4/16/90
OP-005-015	Work Authorization Preparation & Implementation	1	2/7/90
OP-009-005	Shutdown Cooling System	10	3/19/90
OP-009-008	Safety Injection System	8	12/27/89
OP-010-001	General Plant Operations	12	6/1/90
OP-100-001	Duties & Responsibilities of Operators on Duty	6	4/20/90
OP-100-002	Leak Reduction	4	10/7/88
OP-100-003	Caution Tag Control	3	3/9/88
OP-100-009	Control of Valves and Breakers	10	3/31/90
OP-100-007	Shift Turnover	6	6/5/89
OP-100-008	Key Control	3	9/30/89
OP-100-010	Equipment Out of Service	5	2/2/90
OP-500-012	Annunciator Response for Control Room Cabinet N	4	4/20/90
OP-901-004	Evacuation of Control Room & Plant Shutdown	4	6/1/90

OP-901-046	Shutdown Cooling Malfunction	6	3/19/90
OP-902-000	Emergency Entry Procedure	3	8/28/89
OP-902-002	Loss of Coolant Accident Recovery	3	8/28/89
OP-903-008	Reactor Coolant System Isolation Leakage Test	2	5/20/88
OP-903-024	Reactor Coolant System Water Inventory Balance	7	3/31/90
OP-903-026	Emergency Core Cooling System Valve Lineup Verif-ication	4	3/17/89
OP-903-031	Containment Integrity Check	5	6/7/90
OP-903-032	Quarterly IST Valve Test	7	2/28/90
OP-903-033	Cold Shutdown IST Valve Test	8	2/2/90
OP-903-034	Containment Spray Valve Lineup Verification	3	3/16/89

3. Maintenance Procedures

Procedure No.	Title	Rev.	Date
MD-001-011	Maintenance Departmental Procedure Initiation, Review, and Approval of Procedures, Changes, Revision and Deletions; Control and Distribution	5	6/25/90
MD-001-014	Conduct of Maintenance	3	6/30/89
MD-001-016	Failure and Trend Analysis	1	12/16/87
MD-001-028	Writers Guide for Maintenance Department Procedures	2	8/28/89
MD-001-029	Check Valve Monitoring, Maintenance and Trending Program	1	4/1/90
ME-007-008	Motor Operated Valve	8	6/29/90
ME-007-028	MOV Setting, Signature Trend Analysis and Evaluation	0	9/12/89
MI-005-201	Instrument Loop Check	5	7/11/89
MI-005-202	Calibration of Pressure Instruments	3	10/30/85
MI-005-207	Calibration - Indicators	4	10/22/84
MI-005-251	Westinghouse 7300 Card Calibration	4	7/25/89
MI-005-587	Calibration - Pressurizer Pressure	0	7/13/84
MM-006-001	Valve Maintenance	6	1/9/90
MM-006-002	Valve Operator Maintenance	3	3/28/88
MM-006-105	Limitorque Motor Operator Maintenance	2	10/31/86
MM-007-021	Check Valve Monitoring By MOVATS Checkmate System and Inspection	1	4/1/90

4. Miscellaneous

Procedure No.	Title	Rev.	Date
PMD-40	Plant Management Directive Long Term Reliability Program	0	5/10/90
WA-0102859	Valve SI-301 Trouble Shoot		10/25/88
WA-1000663	Valve SI-108B Flange Leaking		8/19/87
WA-1025307	Valve SI-108A Torque Value Monitoring on Valve Flanges		11/21/88
WA-1029002	Valve SI-108B Torque Value Monitoring on Valve Flanges		12/16/88
WA-1013187	Install Isolation Valve Leakoff Line in valve SI-401A		4/20/88
WA-1001923	Cap Packing Gland Leakoff Lines for valves SI-139A, SI-304A, SI-401A&B		6/19/88
WA-1048591	Valve Flange Leakage Calculation for SI-108A		12/7/89
457001173	Lukenhimer Gate Valve Technical Manual	4	10/4/88
	Waterford 3 Pump and Valve Inservice Test Plan 5		
	Waterford 3 Response to Generic Letter 87-06 6/11/87		

5. Drawings

Drawing No.	Title	Rev.
B-424-269S	Pressurizer Pressure Inst.	12E
B-424-515	RCS Hot Leg Injection	13
B-424-530	LPSI Pump A Controls	02
B-424-550	SI Tank 1A Instrumentation	09
B-424 588S	Pressurizer Pressure Inst.	01
B-424-595	SDC Isolation Valves	03
B-424-595S	SDC Isolation Valves	21
B-424-596	SDC Isolation Valves	01
B-424-596S	SDC Isolation Valves	16
B-424-599S	Hydraulic Pump Control	09
B-424-2932	Annunciator Display	08
B-425-319	SI Check Valve Leak Detector	01
B-425-390A	SI HP Pump Controls	01
B-425-390B	SI HP Pump Controls	01
502-13	CP-25 Wiring Diagram	06
503-14	CP-26 Wiring Diagram	06
504-14	CP-27 Wiring Diagram	04
505-14	CP-28 Wiring Diagram	04
506-29	CP-31 Wiring Diagram	05
8821027	CP-50 Wiring Diagram	09
8821038	CP-50 Wiring Diagram	09

6. Piping and Instrumentation Diagrams

Drawing No.	Title	Rev.
LOU-1564 G-167	Safety Injection System (SI) Sheet 1 of 2	28
LOU-1564 G-167	Safety Injection System (SI) Sheet 2 of 2	25
LOU-1564 G-168	Chemical & Volume Control System (CVC), Sheet 1 of 2	25
LOU-1564 G-168	Chemical & Volume Control System (CVC), Sheet 2 of 2	29
LOU-1564 G-172	Reactor Coolant System (RCS)	21
ZSI-700-00	Safety Injection & Shutdown Cooling	T01.03
ZCVC-000-00	CVCS & Boric Acid Makeup Systems	T01.03