ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/96-20
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	August 27-29 with in-office review until October 3, 1996
Inspectors:	L. E. Ellershaw, Reactor Inspector J. Kudrick, Senior Reactor Systems Engineer G. E. Werner, Project Engineer
Approved By:	Dr. Dale A. Powers, Chief, Maintenance Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Partial List of Persons Contacted List of Inspection Procedures Used List of Items Opened, Closed, and Discussed
Attachment 2:	List of Documents Reviewed

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EXECUTIVE SUMMARY

Waterford Steam Electric Station, Unit 3
NRC Inspection Report 50-382/96-20

This inspection was performed using the guidance of NRC Inspection Procedures 73756, "Inservice Testing Of Pumps And Valves," dated July 27, 1995; 92902, "Followup - Maintenance," dated March 14, 1994; and 92903, "Followup - Engineering," dated March 14, 1994, to determine whether the inservice testing program had been properly established and implemented in accordance with Technical Specifications, the applicable ASME Code, correspondence between NRC and the licensee concerning requests for relief and requirements imposed by NRC/industry initiatives.

Maintenance

- Pump est requirements had been established in accordance with the 1980 Edition of the ASME Code through the Winter 1981 Addenda (applicable to Waterford 3), and were being satisfactorily implemented (Section M1.1).

- An inspection followup item was identified to review and evaluate data used to support the licensee's position that safety-related pumps had not degraded to the point where they would not meet system design requirements, and that there were no operability concerns (Section M1.1).

- The failure to shut Containment Spray Valve CS-118A following completion of Train A containment spray pump testing, as required by Step 7.1.23 in Procedure OP-903-035, was an apparent violation of Technical Specification 6.8.1.a (Section M8.1).

- The disposition (over a five month period) of a condition report, which led to the identification of a 10 CFR 50.72(b)(2)(iii) required notification to NRC, was untimely (Section M8.1).

Engineering

- The failure to perform the required written safety evaluation to provide the bases for determining that the difference between the facility design configuration and the Final Safety Analysis Report did not involve an unreviewed safety question was an apparent violation of 10 CFR 50.59 (Section E8.2).

Report Details

Summary of Plant Status

Waterford Steam Electric Station, Unit 3, was at full power operation during this inspection.

II. Maintenance

M1 Conduct of Maintenance

M1.1 ASME Code Section XI Inservice Testing

a. Inspection Scope (73756)

The inspectors reviewed the most recent ASME Code Section XI inservice test data regarding operability verification for selected Train B pumps given in the following procedures:

- "Emergency Diesel Generator Fuel Oil Transfer Pump," Surveillance Procedure OP-903-117, Revision 1;
- "Component Cooling Water Pump," Surveillance Procedure OP-903-050, Revision 11;
- "Boric Acid Pump," Surveillance Procedure OP-903-004, Revision 9;
- "Auxiliary Component Cooling Water Pump," Surveillance Procedure OP-903-050, Revision 11;
- "Charging Pump," Surveillance Procedure OP-903-003, Revision 8;
- "Containment Spray Pump," Surveillance Procedure OP-903-035, Revision 8;
- "Safety Injection Pump," Surveillance Procedure OP-903-030, Revision 10; and
- "Chilled Water Pump", Surveillance Procedure OP-903-063, Revision 9.

b. Observations and Findings

The inspectors determined from their review of the licensee's records that pump test parameters and acceptance criteria had been established in accordance with ASME Code requirements, and that the performance of all pumps met the acceptance criteria.

During their review of the surveillance test package for Component Cooling Water Pump B, the inspectors identified four change notices applicable to the surveillance procedure. One change notice reduced the test flow rate from 6000 to 5600 gpm (Change Notice 1 dated September 18, 1995), and another change notice reduced the test flow rate from 5600 to 4800 gpm (Change Notice 2 dated February 7, 1996). The change from 6000 to 5600 gpm was initiated to provide consistent test conditions, since the surveillance procedure for Component Cooling Water Pump AB specified a flow rate of 6000 gpm, while the surveillance procedures for pumps A and B specified 5600 gpm. The change from 5600 to 4800 gpm was recommended by system engineering to preclude the diversion of component cooling water to the reactor coolant pumps which could cause low flow alarms and increased reactor coolant pump cooler temperatures.

Section IWP-3110 in Section XI of the ASME Code allows a change to reference values, as long as an inservice test at the conditions of an existing set of reference values is performed and the results analyzed. When these are verified as being satisfactory, then a second test is to be performed at the new reference conditions, and these results are then to be used to establish the new set of reference values. The licensee performed these actions and documented the results in accordance with the requirements of Section IWP-3112 in Section XI of the ASME Code. Component Cooling Water Pumps A, B, and AB were each tested using the initial reference values, followed by a test using the new reference conditions. The tests were documented as follows: Component Cooling Water Pump A, Work Authorization 01144519, performed on February 24, 1996; Component Cooling Water Pump B, Work Authorization 01144513, performed on March 15, 1996; and Component Cooling Water Pump AB, Work Authorization 01143621, performed on February 8 and 9, 1996.

The inspectors also learned that licensee personnel had initiated Condition Report CR-96-0414 on March 20, 1996, to address a concern regarding component cooling water pump inservice test acceptance criteria. Licensee personnel had identified that the inservice test acceptance criteria permitted component cooling water pump performance to degrade below the design basis accident performance point without requiring action to restore pump performance. The design flow rate for the component cooling water system was 6554 gpm. The certified pump curve showed that the pump should develop 151 feet of head at 6554 gpm, or 148 feet of head at 6800 gpm. This difference translated to 2 percent of head margin or 3.8 percent of flow margin. However, the inservice test acceptance criteria, which was established in accordance with ASME Code requirements, allowed for (a) 7 percent head degradation and 6 percent pump flow degradation below the reference point, respectively, before the alert range limit was reached, and (b) 10 percent head degradation and 10 percent pump flow degradation before the action range limit was reached. Consequently, this amount of pump degradation

would result in the system performing below the design basis flow rate before the inservice test trend data indicated any problem. The licensee's review of the condition report that discussed this susceptibility resulted in a number of corrective actions, some of which were still open at the end of the onsite inspection.

The licensee performed a review of all other safety-related pumps for similar conditions. Subsequent to the onsite inspection, the inspectors were informed that the review determined that inservice test acceptance criteria, with the exception of the component cooling water pumps and Auxiliary Component Cooling Water Pump A, were set sufficiently high such that pump degradation would be detected before the design basis accident required flow rate was encroached on.

An associated root-cause analysis report dated May 14, 1996, stated that the component cooling water pumps had been, or were, operating in a condition where the pumps would not have provided 148 feet of head at 6554 gpm. It also stated that a previous analysis associated with another condition report (CR 95-0955) determined that even with flow as low as 6000 gpm, the design basis heat load could still be removed by the component cooling water system. In addition, previous special flow balance tests had been performed using Special Test Procedure 01150154, "CCW System Flow Balance Test." The results from those special flow balance tests showed that the total flow rate was approximately 6300 gpm, which was below the design basis accident flow rate of 6554 gpm.

Because of the lower-than-expected flow rates, engineering performed a 10 CFR 50.59 evaluation to assess the impact of the lower flow rates on heat transfer during a design basis accident. The evaluation, approved on October 20, 1995, assumed a 6000 gpm component cooling water system flow rate, which bounded the design flow rate of 6554 gpm and the actual measured flow rate of 6300 gpm. New analyses were performed for the design basis accidents using the assumed lower flow rates. These analyses demonstrated that the component cooling water system heat removal performance with the lower total flow rate was greater than that assumed in the accident analyses. The evaluation concluded that, even with flow rates as low as 6000 gpm, the component cooling water system would still perform its safety function and all applicable acceptance limits would be met.

During their review of the root-cause analysis report, the inspectors noted that the issue of pumps degrading below the design basis accident flow rate before the inservice test program would detect the problem, had been identified at Arkansas Nuclear One as early as 1990. It appeared that in February 1994, Waterford personnel became aware of the issue, and on March 20, 1996, the condition report was issued on this subject. While all corrective action tasks have not been completed, licensee personnel informed the inspectors that their review of all safety-related pump test data for the past year showed that the pumps exceeded the design basis accident flow rates established for their respective systems.

The inspectors did not review and evaluate the data which was used to support the licensee's position that the pumps had not degraded to the point where they would not meet system design requirements, and that there were no operability concerns. This matter will be reviewed as an inspection followup item during a future inspection (50-382/9620-01).

M8 Miscellaneous Maintenance Issues (92902)

M8.1 <u>(Closed) License Event Report 50-382/96-012</u>: Containment Spray Valve CS-118A Found Partially Open.

a. <u>Inspection Scope</u>

The inspectors reviewed Licensee Event Notification 30902 dated August 21, 1996, and Licensee Event Report 50-382/96-012 dated September 20, 1996, which informed the NRC that Containment Spray Valve CS-118A was found partially open (approximately one and one-half turns) on November 11, 1995, subsequent to testing performed on the Train A containment spray pump. Valve CS-118A serves as a flow test line isolation valve. The licensee event notification also identified that during a design basis loss-of-coolant accident while in the recirculation mode, leakage through this valve back to the refueling water storage pool could have exceeded the 8 gpm limit established in Updated Final Safety Analysis Report Table 15.6-19 and, subsequently, could have overexposed the control room operators.

b. <u>Observations and Findings</u>

Following performance of Surveillance Procedure OP-903-094, "ESFAS Subgroup Relay Testing- Operating," Revision 8, licensee personnel observed that containment spray riser level had dropped approximately 55 ft. Investigation revealed that Valve CS-118A, an Anchor Darling, 4 in, normally locked closed, manual gate valve, was not fully closed and was found to be one and one-half handwheel turns open. Condition Report CR 95-1165, which was initiated on November 11, 1995, concluded that containment spray pump operability was not affected since backleakage to the refueling water storage pool was quantified by engineering judgement to be less than the acceptance criterion of 60 gpm.

The system engineering superintendent reviewed Condition Report CR 95-1165 and identified that the condition report failed to address Valve CS-118A backleakage radiological consequences. Consequently, licensing personnel initiated Condition Report CR 96-0287 (March 2, 1996) to address the new issue.

The licensee identified that Valve CS-118A had last been operated on September 19, 1995, during performance of Surveillance Procedure OP-903-035, "Containment Spray Pump Operability Check," Revision 8, Section 7.1, "Containment Spray Pump A." During the surveillance, Valve CS-118A was opened

and then subsequently locked closed; however, discussions with operations personnel revealed that the valve apparently had not been completely closed. This was determined during a subsequent check (as a result of the containment spray riser level drop) on November 11, 1995, when personnel used a "valve operator" (a procedurally allowed leverage bar), and closed the valve an additional one and one-half handwheel turns. Condition Report 95-1165 identified the difficulty in operating the valve as a cause for the valve not being fully closed.

Based upon the above dates, Valve CS-118A was partially open for approximately 53 days; however, the plant was in a refueling outage during this period for 44 days (September 22 to November 5, 1995). In accordance with Technical Specification 3.6.2.1, the containment spray system was not required to be operable in Mode 4 (below 400 psia), and in Modes 5 and 6. The inspectors review of control room logs showed that the plant reached Mode 4 and 350 psia at 0600 on September 23, 1995. Subsequently, during power ascension, the plant reached Mode 4 and 375 psia at 0600 on October 31, 1995. Therefore, the valve was partially open when it was required to be closed between September 19 and 23, and between October 31 and November 11, 1995.

Technical Specification 6.8.1.a, requires that written procedures are to be implemented for those activities referenced in Appendix A of Regulatory Guide 1.33, one of which is surveillances. Surveillance Procedure OP-903-035, used to satisfy that requirement, contained Step 7.1.23, which required that Containment Spray Valve CS-118A be closed and locked following completion of Train A Containment Spray Pump testing. The performance and verification of this step was documented by initials on Step 23 in Attachment 10.1 to the procedure on September 19, 1995. The failure to shut Valve CS-118A as required by Surveillance Procedure OP-903-035 was an apparent violation of Technical Specification 6.8.1.a (50-382/9620-02).

Valve CS-118A was one of the 13 valves previously identified by the licensee as not having been included in the inservice test program and, therefore, not tested (see NRC Inspection Report 50-382/9609). The licensee had placed the valve in the inservice test program, and on March 2, 1996, the valve was tested for leakage. Test results showed no leakage (i.e., 0.0 gpm).

However, from March to June 1996, the licensee attempted to bound the potential leakage problem that existed during September through November 1995, through design calculations, bench testing of a similar valve, and performing a special test through partially opened Valve CS-118A. However, none of the licensee's efforts were successful on quantifying the leakage.

On August 21, 1996, engineering personnel determined they could not quantify refueling water storage pool backleakage at low flow rates. Based upon this information, licensing personnel, in accordance with 10 CFR 50.72(b)(2)(iii), made a 4-hour notification on August 21, 1996. The inspectors reviewed the reportability aspects of this issue and found them acceptable.

Subsequent to the onsite inspection, the licensee submitted Licensee Event Report 50-382/96-012 dated September 20, 1996. In the report, the licensee discussed a special test that was performed on September 7, 1996, in another attempt to quantify the leak rate through Valve CS-118A. While there were no consequences associated with the actual event, the test determined that the maximum leak rate during a design basis loss-of-coolant accident would have been approximately 11.8 gpm, which exceeded the 8 gpm backleakage limit specified in Table 15.6-19 in the Updated Final Safety Analysis Report.

The backleakeage limit was established to maintain the control room thyroid dose limits specified in Criterion 19 of Appendix A to 10 CFR Part 50. This number (11.8 gpm) was arrived at by using actual flow rates from three different valve settings, and by performing valve leakage calculations using expected values during an accident for elevation head, containment spray pump head, valve and line losses, and differential pressure across Valve CS-118A. The test also determined that the backleakage would not have caused off-site dose limits (10 CFR Part 100) to be exceeded. The licensee determined that a leak rate greater than 12.5 gpm during the design basis loss-of-coolant accident would be required to exceed the off-site dose limits. The licensee performed additional control room dose limit calculations. In addition to the excessive backleakage through Valve CS-118A, the licensee considered the number of hours personnel would be in the control room during an accident, and included other possible leakage paths. The Licensee Event Report concluded that the dose attributed to these conditions would not exceed the thyroid dose limit for the control room staff.

c. Conclusions

Pump test requirements had been established and were being implemented in accordance with the 1980 Edition of the ASME Code through the Winter 1981 Addenda (applicable to Waterford 3). An apparent violation of Technical Specification 6.8.1.a occurred when licensee personnel failed to shut Valve CS-118A following a containment spray pump test. The licensee's disposition (over a five month period) of Condition Report 96-0287 (March 2, 1996), which led to the identification of a required report (August 21, 1996) to NRC, was untimely.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.2 Review of Updated Final Safety Analysis Report Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the Updated Final Safety Analysis Report description. While performing the inspections discussed in this report, the inspectors reviewed the applicable sections of the Updated Final Safety Analysis Report that related to the areas inspected.

In addition, the inspectors reviewed the licensee's 10 CFR 50.59 Safety Evaluation Report, "LDCR 97-0047, Location of Cabinets C-3A(B) and C-4 Outside the CVAS Boundary, Rev. 1," dated July 28, 1996. This report was initiated on July 22, 1996, because the licensee became aware that their justification, in response to Final Safety Analysis Report Question 480.36, for not conducting 10 CFR Part 50, Appendix J, Type C leak tests on the valves in the instrument lines through Containment Penetrations 53 and 65, was not correct (Appendix J, Type C leak tests are pneumatic tests used to measure containment isolation valve leakage rates). This resulted in errors between the wording of the Updated Final Safety Analysis Report and the current plant configuration regarding the containment vacuum relief lines.

The 10 CFR 50.59 report identified the following Updated Final Safety Analysis sections and tables requiring changes: Section 1.9.37; Section 7.1.2.7; Table 3.9-9; Table 6.2-32, and Table 6.2-43. In addition, the following tables in the Technical Requirements Manual were also identified as requiring changes: Tables 3.6-1 and 3.6-2. Licensee personnel stated that the Updated Final Safety Analysis Report would be changed to reflect the actual plant configuration, and to revise leak rate test requirements.

Additional details and information pertaining to the potential consequences of these discrepancies are located below in Sections E8.1 and E8.2.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-382/9609-04: the determination of whether there were active safety functions for Excess Flow Check Valves CVR-302A(B), "Containment/Annulus Differential Pressure Sample Line Check Valve."

Containment Penetrations 53 and 65 each contain two containment vacuum relief instrument lines. One instrument line senses differential pressure across the containment vessel and provides a signal to actuate the containment vacuum relief

system. The other instrument line monitors the same differential pressure and provides an input to the plant computer.

Excess Flow Check Valves CVR-302A(B) were installed in the containment vacuum relief sensing instrument lines to meet the guidance specified in Regulatory Guide 1.11. This was committed to in Section 6.2.4.2.2 of the Updated Final Safety Analysis Report. Regulatory Guide 1.11 states, in part, "In the event of a rupture downstream of the valve, the valve should close automatically or be capable of being closed during normal reactor operation and under accident conditions."

In Condition Report 96-0272, dated March 1, 1996, the licensee identified that Valves CVR-302A(B) had a closed safety function. They were added to the inservice test plan as ASME Code Category C valves, but with a cold shutdown test frequency justification[1]. The condition report further stated that these valves were listed in Table 3.6-2 of the Technical Requirements Manual which identified them as containment isolation valves and subject to Technical Specification 3.6.3, and 10 CFR Part 50, Appendix J testing.

The licensee's representatives indicated that the valves had been tested during previous refueling outages in the closed direction and were capable of meeting their closed safety function. However, these penetrations were only tested during the 10 CFR Part 50, Appendix J, Type A tests. (Appendix J, Type A tests measure the containment system overall integrated leakage rate under conditions representing design basis loss-of-coolant accident containment peak pressure). No individual testing of the valves was performed. Licensee personnel informed the inspectors that closed-system leak tight integrity tests had been conducted on the closed systems during refueling outages; however, this required that the valves be open in order to verify system integrity.

Subsequently, licensee personnel stated that, following additional reviews, they identified that the excess flow check valves did not have a closed safety function and did not have to be tested in the closed direction. The licensee supported this position by referencing Final Safety Analysis Report Question and Response 480.36 and Updated Final Safety Analysis Report, Table 6.2-32, which showed that 10 CFR Part 50, Appendix J, Type C containment isolation valve leak testing was not required for these penetrations. In addition, Table 6.2-43, showed that Penetrations 53 and 65 were tested during Appendix J, Type A tests and were not required to have a separate Appendix J, Type C test. However, NRC acceptance of that position, as stated in the Safety Evaluation Report, "Waterford Steam Electric

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[1]They also were part of the subject of apparent violation 50-382/9609-02, regarding a failure to include valves in the inservice test program that were required to be tested in accordance with Section XI of the ASME Code.

Station, Unit 3, NUREG-0787" dated July 1981, did not imply that no testing was acceptable. It simply indicated that an Appendix J, Type C leak rate test was not necessary.

On July 20, 1996, while the plant was in Mode 5, Cold Shutdown, the licensee performed an inservice stroke test on Valve CVR-302B, and the valve failed to close. The valve was declared inoperable, and Condition Report CR-96-1103 was initiated. On July 21, 1996, a stroke test was performed on Valve CVR-302A, with similar results. The failure of Valve CVR-302A was incorporated into the existing condition report.

The failed valves were removed from the containment vacuum relief sensing lines. The replacement valves were tested and found to be acceptable. The licensee's evaluation concluded that had these valves failed during plant operations, there would not have been a safety concern. This evaluation was based on the fact that the tubing downstream of these valves was safety related (i.e., fabricated to ASME Code, Class 3 requirements, and classified as Seismic Category 1), and constituted a closed system outside of containment which would be assumed to fulfill its intended function (i.e., maintain its integrity) under accident conditions. The inspectors agreed with the licensee's conclusion.

During the assessment of Condition Report CR 96-1103, the NRC inspector discussed the issue with licensee personnel, who indicated that the instrument lines did not terminate in the controlled ventilation area system, and that a potential for bypass leakage existed. The inspectors, on July 21, 1996, questioned licensing personnel about what appeared to be a discrepancy, in that part of the licensing basis was predicated on the lines terminating in the controlled ventilation area system. On July 22, 1996, further inspector questioning caused licensee personnel to conduct a review of their response to Final Safety Analysis Report Question 480.36 regarding justification for not conducting Appendix J, Type C leak tests on the valves in the instrument lines through Containment Penetrations 53 and 65.

Through this review, the licensee became aware that an incorrect response had been provided to the NRC. This caused the initiation of Condition Report CR 96-1123 dated July 23, 1996, which documented this discrepancy. On August 26, 1996, Licensee Event Report 50-382/96-009, which is discussed below, was issued to address the issues identified during the licensee's review. Subsequently, an August 12, 1996, predecisional enforcement conference was postponed pending review of the issues that were raised by the licensee's report.

E8.2 (Closed) LER 50-382/96-009: failure of Containment Vacuum Relief Valves-402A(B) (excess flow check valves). On July 22, 1996, the licensee became aware that the cabinets for the instrumentation associated with the containment vacuum relief system (sensing and monitoring lines) were not within the controlled ventilation area system, and that this discrepancy was inconsistent with the licensing basis. Previously, on July 5, 1996, the licensee determined that

the required Technical Specification testing had not been performed on Valves CVR-401A(B) and CVR-402A(B) which were located in the containment vacuum relief monitoring lines.

During initial licensing of Waterford Steam Electric Station, the licensee responded to Final Safety Analysis Report Question 480.36 (dealing with testing of certain containment penetrations) by stating that the containment vacuum relief sensing lines and the containment vacuum relief monitoring lines each formed a closed system outside of containment, were seismically qualified, and terminated within an area exhausted by the controlled ventilation area system (a filtered ventilation system). Based on this rationale, the NRC staff determined that the design and isolation provisions of Containment Penetrations 53 and 65 would be acceptable to meet General Design Criterion 56 of Appendix A to 10 CFR Part 50. This determination was included in Section 6.2.4 of Safety Evaluation Report, "Waterford Steam Electric Station, Unit 3, NUREG-0787" dated July 1981.

Further licensee review of their response revealed that, not only did the instrument lines not terminate in the controlled ventilation area system, but the monitoring lines did not meet the design criteria for a closed system outside of containment and they were not seismically qualified.

10 CFR 50.59(a)(1) allows the holder of a license to make changes to the facility as described in the safety analysis report unless the proposed change involves an unreviewed safety question. 10 CFR 50.59(b)(1) requires the licensee to maintain records of changes in the facility, to the extent that these changes constitute changes in the facility as described in the safety analysis report. The records must include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

The actual design configuration of the containment vacuum relief system was different from that described in the licensee's response to Final Safety Analysis Report Question 480.36, and until July 28, 1996, the licensee failed to perform the required written safety evaluation to provide the bases for a determination that the deviation from the response to Final Safety Analysis Report Question 480.36 did not involve an unreviewed safety question. This failure was an apparent violation of 10 CFR 50.59 (50-382/9602-03).

Criterion 56 of Appendix A to 10 CFR Part 50 requires each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with two containment isolation valves unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis. In addition, the use of a simple check valve as the automatic isolation valve outside containment is not allowed.

Regulatory Guide 1.11, which the licensee committed to in Section 6.2.4.2.2 of the Updated Final Safety Analysis Report, provided suitable bases for demonstrating acceptable alternative containment isolation provisions. However, in order to use the provisions of Regulatory Guide 1.11, the valving provided for each instrument line penetrating containment must reflect the importance of two safety features: (1) the function the line performs, and (2) the need to maintain containment leaktight integrity.

With respect to the sensing lines, the importance of the safety function was recognized and application of Regulatory Guide 1.11 was justified. The lack of a safety function for the monitoring line, however, established that use of Regulatory Guide 1.11 was not an appropriate alternative with respect to the isolation provisions as described in General Design Criterion 56, and the use of an excess flow check valve was not an acceptable isolation barrier. The failure to implement General Design Criterion 56 requirements by not providing the specified containment isolation barriers in the containment vacuum relief monitoring lines was a result of the licensee failing to recognize that the actual design configuration was different from their response to Final Safety Analysis Report Question 480.36.

The inspectors also noted that during the operating license review, the NRC staff asked for additional justification for not including Containment Penetrations 53 and 65 in an Appendix J, Type C leak testing program as discussed in Question 480.36. The licensee response reiterated the isolation barriers as described above, but also indicated that Solenoid Operated Valves CVR-401A(B), which would close on a containment isolation signal, would be added to the containment vacuum relief monitoring lines during the first refueling outage. There was no mention of any testing requirements associated with those valves since the acceptance of the penetrations was based on the premise that closed systems existed in conjunction with the excess flow check valves. The licensee's representatives stated that the valves would be open for each Appendix J, Type A test so that the integrity of the closed system would be tested. In addition, licensee personnel informed the inspectors during this inspection, that a local leak tight integrity test had been conducted on all the closed systems during those refueling outages when the Appendix J, Type A test was not performed.

After identifying that the containment vacuum relief monitoring lines did not constitute a closed system, the licensee isolated these lines by closing the upstream Solenoid Isolation Valves CVR 401A(B). On July 26, 1996, Valves CVR-401A(B) and CVR-402A(B) were leakrate tested under Work Authorization 01149602 with the following results, as given in Table 1.

TABLE 1

VALVE	AS-FOUND LEAKRATE (sccm)	AS-LEFT LEAKRATE(sccm)
CVR-401A	20.0	20.0
CVR-401B	21.6	21.6
CVR-402A	129,000	44500*
CVR-402B	242,500	44500*

* Assigned design leakage - actual measured leakage was less than this value.

Excess Flow Check Valves CVR-402A(B) failed to close during the initial leakrate test and were removed. It appeared that these valves failed because of excessive manufacturing roughness in the valve body bore. The replacement valve body bores were smoothed prior to installation and were successfully tested as shown above. Technical Specification 3.6.1.2 limited combined bypass leakage to a rate of less than or equal to 0.06 L_a. The Solenoid Isolation Valves CVR-401A(B) were left shut since the combined bypass leakage through these lines could exceed 0.06 L_a if both solenoid valves failed to shut. This leakage was determined by licensee calculation to be equal to 63,069 sccm and was incorporated into Surveillance Procedure STA-001-006, "Leak Rate Testing," Revision 2, Attachment 10.9. As of July 26, 1996, with Valves CVR-401A(B) shut, the combined bypass leakage was 14,924.7 sccm.

Technical Specification Surveillance Requirement 4.6.1.2.d required that bypass flow be measured at least once per 24 months by either performing a Type B (not applicable to containment isolation valves) or Type C test on individual penetrations that can be tested. The failure of the licensee to test Penetrations 53 and 65 prior to July 26, 1996, in accordance with Technical Specification Surveillance Requirement 4.6.1.2.d, was another result of the licensee not recognizing the discrepancy between the actual design configuration of the containment vacuum relief system and their response to Final Safety Analysis Report Question 480.36.

The licensee determined that both control room doses and offsite doses could have been exceeded had a design basis loss-of-coolant accident occurred assuming rupture of the nonessential monitoring line and failure of either Valve CVR-401A or CVR-401B with Valves CVR-402A or CVR-402B failed in the open position (as-found condition). Licensee calculated control room and offsite doses are shown in Table 2.

TABLE 2

LOCATION	THYROID DOSE (rem)	WHOLE BODY DOSE (rem)	SKIN DOSE (rem)
2 Hr Exclusion Area Boundary	405.82 (300)*	13.43 (25)*	N/A
30 Day Low Population Zone	162.85 (300)*	3.323 (25)*	N/A
30 Day Control Room	45.71 (30)*	1.36 (5)*	31.51 (30)*

* The values within parentheses are regulatory limits from the standard review plan.

On August 21, 1996, the licensee submitted License Amendment Request W3F1-96-01441, "Discrepancy Regarding the Design and Testing of Instrument Sensing Lines Penetrating the Primary Containment," to resolve the design and testing discrepancies associated with both the containment vacuum relief sensing and containment vacuum relief monitoring lines.

The inspectors reviewed the Updated Final Safety Analysis Report for other instrumentation lines penetrating containment and reviewed the construction of those lines. The inspectors verified that no other containment penetration lines utilized excess flow check valves. Also, the licensee searched their parts data base and identified that the excess flow check valves were used exclusively on the Containment Vacuum Relief System (Penetrations 53 and 65). Updated Final Safety Analysis Report Section 6.2.4.2.2, "Instrument Lines," identified only one additional instrument line (Penetration 54) that penetrated containment.

This instrument line was used for the wide range containment pressure instrumentation, and was depicted on Drawing B-430, Sheet P-77. The Updated Final Safety Analysis Report described the containment wide range pressure instrumentation as consisting of a sealed, liquid-filled system with a bellows, constructed in accordance with ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems." ANSI N271-1976 was used to satisfy the requirement of 10 CFR Part 50, Appendix A, Criterion 56.

The inspectors performed a walkdown of the piping and instrumentation outside containment for Penetration 54 and identified that the transmitter was not enclosed in protective shielding (protection from missiles and water jets) as required by ANSI N271-1976. Followup discussion with the licensee identified that the piping and transmitter were located in an area that had been analyzed for jet impingement and

no jet impingement concerns were identified. In addition, other components located in the same area were all seismically qualified; therefore, no missiles would be associated with their failure. Based upon this information, the inspectors determined that Penetration 54 hardware were adequate to provide the required safety function.

V. Management Meetings

X1 Exit Meeting Summary

Subsequent to the conclusion of the inspection on October 3, 1996, the inspector telephonically presented the inspection results to members of licensee management on October 9, 1996. The licensee acknowledged the findings presented.

The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Caropino, Licensing Coordinator
C. Dugger, General Manager, Plant Operations
J. Fisicaro, Director, Nuclear Safety
C. Fugate, Shift Supervisor, Operations
T. Gaudet, Manager, Licensing
P. Gropp, Supervisor, Mechanical Specialties, Design Engineering
J. Holman, Manager, Safety Analysis
J. Houghtaling, Technical Assistant, Design Engineering
J. Howard, Manager, Procurement/Programs Engineering
P. Melancon, Inservice Test Engineer
G. Robin, Supervisor, Programs Engineering
L. Rushing, Manager, Mechanical/Civil Design Engineering
M. Selman, Vice President, Operations
P. Stanton, Design Engineer, Mechanical Systems
C. Thomas, Supervisor, Licensing
R. Thweatt, Supervisor, Design Engineering, Mechanical
D. Urciuoli, Licensing Engineer
D. Vinci, Manager, Systems Engineering
K. Walsh, Lead Senior Engineer, Operations
A. Wrape III, Director, Design Engineering

NRC

K. Brockman, Acting Director, Division of Reactor Safety
D. Powers, Chief, Maintenance Branch
L. Keller, Senior Resident Inspector
T. Pruett, Resident Inspector

LIST OF INSPECTION PROCEDURES USED

IP 73756	Inservice Testing of Pumps and Valves
IP 92902	Followup - Maintenance
IP 92903	Followup - Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-382/9620-01	IFI	An inspection followup item was identified to review and evaluate data used to support the licensee's position that safety-related pumps had not degraded to the point where they would not meet system design requirements and that there were no operability concerns.
50-382/9620-02	EEI	The failure to shut Containment Spray Valve CS-118A following completion of Train A containment spray pump testing, as required by Step 7.1.23 in Procedure OP-903-035, was an apparent violation of Technical Specification 6.8.1.a.
50-382/9620-03	EEI	An apparent violation regarding the failure to perform the required written safety evaluation to provide the bases for a determination that the difference between the facility design configuration and the Final Safety Analysis Report was not an unreviewed safety question. This resulted in a failure to meet General Design Criterion 56 requirements with respect to providing the specified containment isolation barriers in the containment vacuum relief system, and a failure to test Penetrations 53 and 65 prior to July 26, 1996, in accordance with Technical Specification Surveillance Requirements.

Closed

50-382/9609-04	URI	This item pertained to determination of active safety functions for excess flow check valves CVR-302 A(B).
50-382/96-009	LER	This item dealt with failure of Containment Vacuum Relief Valves-402A(B) (excess flow check valves).
50-382/96-012	LER	This item dealt with the failure to close Containment Spray Valve CS-118A following pump testing.

ATTACHMENT 2

LIST OF DOCUMENTS REVIEWED

Design Documents

"Waterford Quality Assurance Program Manual," Revision 5

"Technical Specifications," Amendment 112

"Updated Final Safety Analysis Report," Revision 9

"Waterford III Pump and Valve Inservice Test Plan," Revision 8, Change 1

"Waterford III Pump and Valve Inservice Test Plan," Revision 7, Change 10

W3-DBD-024, "Inservice Testing Basis Document," Revision 0 and Revision 1

W3-DBD-014, "Safety-Related Air-Operated Valves," Revision 0

W3-DBD-042, "Sampling System," Revision 0

Procedures

Surveillance Procedure OP-903-035, "Containment Spray Pump Operability Check," Revision 8

Surveillance Procedure OP-903-117, "Emergency Diesel Generator Fuel Oil Transfer Pump," Revision 1

Surveillance Procedure OP-903-050, "Component Cooling Water Pump," Revision 11

Surveillance Procedure OP-903-004, "Boric Acid Pump," Revision 9

Surveillance Procedure OP-903-050, "Auxiliary Component Cooling Water Pump," Revision 11

Surveillance Procedure OP-903-003, "Charging Pump," Revision 8

Surveillance Procedure OP-903-035, "Containment Spray Pump," Revision 8

Surveillance Procedure OP-903-030, "Safety Injection Pump," Revision 10

Surveillance Procedure OP-903-063, "Chilled Water Pump," Revision 9

Surveillance Procedure STA-001-006, "Leak Rate Testing," Revision 2

Administrative Procedure UNT-006-010, "Event Notification and Reporting," Revision 14

Administrative Procedure, UNT-006-011, "Condition Report," Revision 4

Administrative Procedure, UNT-006-021, "Pump And Valve Inservice Testing," Revision 2

Site Directive, W2.302, "10 CFR 50.59 Safety and Environmental Impact Screening and Evaluation"

<u>Condition Reports</u>

CR 95-1165

CR 96-0287

CR 96-0272

CR 96-0414

CR 96-1123

<u>Work Authorizations</u>

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