UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

March 30, 2000

Charles M. Dugger, Vice President
Operations - Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066-0751

SUBJECT: SUMMARY OF PUBLIC MEETING HELD ON FEBRUARY 22, 2000, AND NRC INSPECTION REPORT NO. 50-382/00-01

Dear Mr. Dugger:

This letter documents the meeting held on February 22, 2000, in the Region IV office in Arlington, Texas, to discuss the method that you employ to account for instrument uncertainties in safety-related systems. A list of attendees is included as Enclosure 1 and a copy of your handouts is included as Enclosure 2. This letter also refers to the inspection conducted on February 28 through March 3, 2000, at the Waterford Steam Electric Station, Unit 3, facility. The enclosed report presents the results of this inspection.   Also, a followup exit was conducted via teleconference with your staff on March 30, 2000, to inform you of the reclassification of certain issues that were presented at the exit meeting on March 3, 2000.

The public meeting was beneficial to NRC in attaining an understanding of your method of accounting for instrument uncertainties for safety-related systems. The information provided by your personnel aided our focusing more expeditiously on the issues and reaching an assessment of your process during the subsequent inspection.

The primary focus of the inspection was an unresolved item that was associated with the potential that you could not demonstrate the ability of several safety-related pumps to provide adequate flow when instrument uncertainties were considered. The results of the inspection show that you were capable of demonstrating the ability of the subject pumps to provide the required flows during accident conditions when appropriate instrument uncertainties were applied, either quantitatively or qualitatively. The remainder of the inspection involved review of other previously identified issues.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as noncited violations, consistent with Section VII.B.1.a of the Enforcement Policy. These noncited violations are described in the subject inspection report. If you contest the violation or severity level of these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, the Director, Office of Enforcement, United States Nuclear

Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Waterford Steam Electric Station, Unit 3, facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its 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

Dr. Dale A. Powers, Acting Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Docket No.: 50-382
License No.: NPF-38

Enclosure: NRC Inspection Report No.
50-382/00-01

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-382
License No.:	NPF-38
Report No.:	50-382/00-01
Licensee:	Entergy Operations, Inc.
Facility:	Waterford Steam Electric Station, Unit 3
Location:	Hwy. 18 Killona, Louisiana
Dates:	February 28 through March 3, 2000
Inspector:	C. J. Paulk, Senior Reactor Inspector Engineering and Maintenance Branch
Accompanied By:	R. W. Deese, Reactor Inspector Engineering and Maintenance Branch D. Prevatte, Contractor Beckman & Associates
Approved By:	Dr. Dale A. Powers, Acting Chief Engineering and Maintenance Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	List of Attendees at Public Meeting Held February 22, 2000, in Arlington, Texas
Attachment 3:	Licensee Handout at Public Meeting on February 22, 2000

EXECUTIVE SUMMARY

Waterford Steam Electric Station, Unit 3
NRC Inspection Report No. 50-382/00-01

This reactive inspection was performed, primarily, to address an unresolved item associated with the application of instrument uncertainties when demonstrating the capability of safety-related pumps to perform their design basis functions. The inspection involved two inspectors and one contractor for one week.

The results of the inspection indicate that the subject safety-related pumps were capable of providing adequate flows under design basis conditions when appropriate instrument uncertainties were accounted for, either quantitatively or qualitatively.

Engineering

- A noncited violation of Criterion III of Appendix B to 10 CFR Part 50 was identified for the failure to assure single failure protection (i.e., separate power supplies) for the hydrogen analyzer containment isolation valves. This issue was discussed in Licensee Event Reports 50-382/97032-00 and 50-382/97032-01 (Section E8.1).

- A noncited violation of Technical Specification 6.8.1a. was identified for failure to identify the need for periodic replacement of elastomeric components in safety-related snubbers. This was documented in the licensee's corrective action program as Condition Report CR-WF3-2000-0192 (Section E8.6).

Report Details

Summary of Plant Status

The plant was operated at 100 percent power during the inspection.

III. Engineering

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Licensee Event Reports 50-382/97032-00 and 50-382/97032-01: Hydrogen analyzer piping did not meet the requirements for reliable and redundant containment isolation valves.

This issue was discussed in NRC Inspection Report 50-382/97-25. No new issues were revealed by the event reports. Based on review of the previous inspection report and the event reports, the inspectors concluded the following.

10 CFR Part 50, Appendix B, Criterion III, states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Section 3.1.47 of the Final Safety Analysis Report, "Criterion 54 - Piping Systems Penetrating Containment," states that the design bases require a double barrier on all piping penetrating the containment vessel so that no single failure or malfunction of an active component can result in loss of isolation or intolerable leakage.

Contrary to the above, the licensee did not assure, from the time of the initial design to the time of correction, that measures were adequate to correctly translate into specifications the design bases to provide a single failure-proof containment isolation for the hydrogen analyzer system. This nonrepetitive, licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-382/0001-01).

E8.2 (Closed) Licensee Event Report 50-382/97033-00: 10 CFR 50.46 acceptance criterion exceeded procedure.

This issue was discussed in NRC Inspection Report 50-382/97-25. No new issues were revealed by the event report. This event has been characterized as Example 1 of Violation D of Enforcement Action 98-022, discussed below.

E8.3 (Closed) Licensee Event Report 50-382/98017-00: Calculated peak cladding temperature exceeded 10 CFR 50.46 criterion.

On July 12, 1997, Combustion Engineering InfoBulletin 97-04 identified an error in the energy redistribution factors used in the Waterford-3 loss-of-coolant accident analysis. The identified energy redistribution factor error, when applied to the loss-of-coolant accident analysis, resulted in a calculated peak cladding temperature greater than the 10 CFR 50.46(b)(1) criterion of 2200°F (1204.4°C). To ensure immediate compliance to the 2200°F (1204.4°C) limit, a 0.2 kW/ft penalty for planar linear heat generation rate was added to the core operating limit supervisory system. A historical search found that the plant was never operated within 0.2 kW/ft of conditions that would have challenged the peak cladding temperature limit during the limiting case accident. As a result, the calculated peak cladding temperature never exceeded the limit.

E8.4 (Closed) Enforcement Action 50-382/98022, Violation C: Untimely and ineffective corrective actions (4 examples).

The inspectors verified the corrective actions described in the licensee's response letter, W3F1-98-0119, dated July 29, 1998, to be reasonable and complete. No problems were identified.

E8.5 (Closed) Enforcement Action 50-382/98022, Violation D: Surveillance procedure did not include provisions for assuring that adequate test instrumentation was used (2 examples).

The inspectors verified the corrective actions described in the licensee's response letter, W3F1-98-0119, dated July 29, 1998, to be reasonable and complete. No problems were identified.

E8.6 (Closed) Unresolved Item 50-382/9904-02: Degraded steam generator hydraulic snubbers.

As documented in NRC Inspection Report 50-382/99-04, on February 26, 1999, licensee personnel found that Snubbers SG-MSNB-734-1A, SG-MSNB-735-1A, and SG-MSNB-736-1A, attached to Steam Generator 1, had empty hydraulic oil reservoirs. No problem with snubbers attached to Steam Generator 2 were identified. Also, the licensee personnel noted the presence of hydraulic oil on the structural steel below the hydraulic reservoirs for all three snubbers. Upon further investigation into this anomaly, the licensee personnel discovered an air void in the snubber assemblies. The inspector questioned whether the condition would allow for hydraulic dampening in a seismic event and, if not, whether the snubbers should be considered inoperable. The licensee generated Condition Report CR-WF3-1999-0212 to document and resolve the condition.

Licensee personnel had taken corrective actions as a result of the inspectors identifying the lack of hydraulic fluid in the reservoirs. They determined the source of the leak to be the hoses connecting the snubber assemblies to their reservoirs. Maintenance personnel replaced the hoses and refilled the snubbers up to a visible level in the reservoirs. Also, the licensee's engineers revised the snubber visual inspection frequency from every other refueling outage to every outage of sufficient length that occur more than 60 days after a previous outage. Further research into the cause of the leaking hoses led the licensee's engineers to conclude that they were not assuring that the qualified life of the safety-related components established by the vendor was being met by not replacing the hoses and other elastomeric components every 5 years as set forth in Vendor Manual TM-E250.0305, "Vendor Technical Manual for Enertech Hydraulic Shock Suppressors," Revision 5. Section C, "Preventative [sic] Maintenance Instructions," Part 1.0, states to replace the rod wiper, the chevron pack, and the wave spring on a 5-year basis, as a result of possible radiation damage.

The failure to implement the requirement to replace the age-sensitive hoses and elastomeric components is a violation of Technical Specification 6.8.1a, which states that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, Appendix A, lists typical safety-related activities that should be covered by written procedures. Included in this list are procedures for performing maintenance. Specifically, Step 9.a of Appendix A states, in part, that "[m]aintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures . . ." Step 9.b states, in part, that "[p]reventive maintenance schedules should be developed to specify . . . inspection or replacement of parts that have a specific lifetime . . ."

This nonrepetitive, licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-382/0001-02). This condition was documented in the licensee's corrective action program as Condition Report CR-WF3-2000-0192. As a result of the finding, the licensee adopted a replacement plan for hoses and other elastomeric components of every third refueling outage.

The licensee's staff demonstrated a lack of knowledge of the snubbers' construction and operation as evidenced by the manner they collected hydraulic oil from the snubbers. The inspectors observed a number of actions taken by licensee personnel that contributed to an inaccurate determination of the oil lost from the snubbers.

First, for only one snubber, the inspectors noted that the maintenance personnel had recorded the amount of oil added. However, the level the reservoir was filled to was not annotated. Had the final reservoir level been noted, the exact volume of the air void could have been determined for that snubber.

Then, for the other two snubbers, the inspectors noted that the maintenance personnel determined, from a visual estimation, that the distance between the inside diameter of the snubber to the oil level was 1.5 to 2 in (3.81 to 5.08 cm). Any direct measurement of the level (e.g., a dipstick) could have quantified the oil level more precisely and led to a more accurate determination of oil content in those snubbers. Next, the oil was collected in collapsible buckets, which contributed to an inaccurate measurement of the amount of oil remaining in these snubbers. Finally, oil was also spilled during the collection process from at least one of the two snubbers and not all of the oil was recovered, i.e., there was still oil in the snubber when the draining stopped. The inspectors noted that the licensee's engineers assumed that exactly the same amount of oil was spilled for both snubbers.

E8.7 (Closed) Unresolved Item 50-382/9906-04: Application of instrument uncertainties.

During this inspection, the inspectors reviewed issues associated with the accounting for instrument uncertainties that were identified as an unresolved item in NRC Inspection Report 50-382/99-06. This unresolved item resulted from a review of Unresolved Item 50-382/98201-18 that was identified during an architect-engineering inspection.

The unresolved item that was previously reviewed and closed involved the accuracies of instruments used in the performance of ASME Code, Section XI, "Inservice Testing on Safety-Related Pumps," which is required by the Updated Final Safety Analysis Report, Section 3.9.6, Technical Specification 4.0.5, and 10 CFR 50.55a(g). This testing is also required to be performed in accordance with ASME/ANSI OMa-1988, (Part 6), "Inservice Testing of Pumps in Light-Water Reactor Power Plants." This standard requires, in part, that instrument accuracy be $\leq \pm 2$ percent of full scale. However, the architect-engineering inspection team discovered that the instruments used for the surveillance tests on the auxiliary component cooling water, component cooling water, low pressure safety injection, high pressure safety injection, and chemical and volume control pumps did not meet this requirement. Subsequent review of the actual test data by the architect-engineering followup inspection team confirmed that, with allowance for the actual error of the instruments used, the pumps' performances were within the ASME Code limits. On that basis, the unresolved item was closed.

In reviewing that unresolved item, however, the architect-engineering followup inspection team determined that, although the revised procedures required the pumps to meet ASME Code performance limits (typically, 10 percent degradation from reference performances), they did not address the ability of the pumps to meet their design basis performance limits, which could be more restrictive than ASME Code limits. (This NRC generic concern had been documented in NRC Information Notice 97-90, "Use of Nonconservative Acceptance Criteria in Safety-Related Pump Surveillance Tests.")

Licensee engineers responded that current test data indicated that all pumps in the inservice testing program also met their respective design basis performance limits. However, the architect-engineering team identified that, in making this determination, the licensee's engineers had not applied the instrument uncertainties to the test data. Therefore, there was a potential that, when these uncertainties were applied, the pumps would not meet their design basis performance limits. This issue was related to an ongoing enforcement action (EA 98-022) identified in NRC Inspection Report 50-382/97-25. In response to the team's concern, licensee engineers generated Engineering Request ER-W3-99-0428, dated April 22, 1999.

At the conclusion of the architect-engineering followup inspection, further NRC evaluation was necessary to determine the ability of the high pressure safety injection, auxiliary component cooling water, component cooling water, chemical and volume control, essential chilled water, and emergency feedwater pumps to provide adequate performance to meet their design basis requirements.

During a public meeting held December 2, 1999, in NRC headquarters, licensee personnel presented information to the staff to support their position that instrument uncertainties did not have to be explicitly addressed. This presentation was made after the licensee received correspondence from the Office of Enforcement documenting NRC review of the licensee's denial of the violations identified in Enforcement Action 99-022, dated August 18, 1999. In that letter, NRC stated that "the [c]onservatisms inherent in the [10 CFR Part 50] Appendix K methodology do not envelop emergency core cooling system flow uncertainties. Such uncertainties must be accounted for and this can be done either through the analyses itself or through the surveillance program. Instrument uncertainties must be managed in a manner that ensures that technical specification limiting conditions for operation (LCOs) preserve the analytical values on which the LCOs are based."

When presenting the position that the uncertainties could be implicitly accounted for, the licensee's representatives were not able to answer all of the staff's questions. In order to come to resolution for this issue, this inspection was performed. The licensee's general response to this concern was that, in every case, surveillance testing instrument uncertainties were accounted for in the pumps' respective accident analyses, either explicitly or implicitly. More specifically, for the high pressure safety injection pumps, explicit accounting was made in the accident analyses, and for the other pumps, the available analysis margins were large and adequate to accommodate these uncertainties. The inspectors asked the licensee's engineers to provide specific evidence and discussions to verify these claims. The following are the results of these discussions and the inspectors' review:

<u>High Pressure Safety Injection Pumps</u>

As referenced above, the engineering team inspection described, in NRC Inspection Report 50-382/97-25, that although the high pressure safety injection pump test instrument uncertainty had been accounted for in the accident analysis, only 5 gpm [18.9 Lpm] had been included in the analysis; whereas, the actual uncertainty as determined by licensee analysis was 18 gpm [68.1 Lpm]. With the reduction in high pressure safety injection flow that this additional uncertainty could represent, there was the potential that the 10 CFR 50.46 limit on the design basis loss-of-coolant accident peak cladding temperature (2200°F [1204.4°C]) could be exceeded.

In response to this finding, licensee personnel generated Calculation EC-195-011, "SI-HPSI Flow Instrumentation Loop Uncertainty Calculation," Revision 2, to account for all high pressure safety injection pump testing uncertainties. This calculation identified 14.1 gpm [53.4 Lpm] of instrument uncertainty plus a 5 gpm [18.9 Lpm] bias factor per injection leg for a total of 19.1 gpm [72.3 Lpm] per injection leg (three legs). It also identified 12.3 gpm [46.6 Lpm] of uncertainty per leg associated with the throttle valve position. By applying the square root of the sum of the squares methodology to the random factors and algebraically adding the bias factor, these errors were combined in the response to Engineering Request ER W3-98-0009. The licensee's engineers substituted that result for the original instrument uncertainty allowance to obtain new high pressure safety injection accident flow rates, which were then used in a revised accident analysis. This change, coupled with changes in the analyses model produced a peak cladding temperature of 1929°F [1053.9°C] (previously 1879°F [1026.1°C]). Therefore, the appropriate instrument uncertainty was incorporated into the accident analyses whose results met the requirements of 10 CFR 50.46.

It should also be noted that additional margin was demonstrated by Calculation EC-M98-069, "HPSI System Performance Surveillance Requirements Basis," Revision 0. The inspectors determined that this calculation showed, for injection conditions, there was a margin of 98 gpm [371 Lpm] between the 583 gpm [2206.9 Lpm] used in the accident analysis and the 675 gpm [2555.2 Lpm] technical specification limit.

Therefore, the inspectors concluded that adequate margin existed to account for instrument uncertainties.

Chilled Water Pumps

The inspectors noted that the worst-case event for the chilled water system was a design basis tornado. For this event, the design basis analysis chilled water pump flow limit was 461.5 gpm [1747 Lpm]. In accordance with the technical specification surveillance test acceptance criteria, at the 10 percent degraded condition allowed by ASME Section XI, the minimum allowable flow was 500 gpm [1892.7 Lpm]. At this condition, there still existed an 8.3 percent performance margin. The licensee's engineers also stated that the cooler with the least flow margin to meet its design basis heat transfer requirements, the room cooler for the emergency feedwater Pump A, had a performance margin of 9 percent. The inspectors concluded that the combination of these considerations provided adequate margin for pump test instrument uncertainties.

Component Cooling Water Pumps

For the component cooling water pumps, the licensee's engineers maintained that, although instrument uncertainty was not explicitly included in the surveillance test acceptance criteria or in the supporting analyses, there were ample margins in the analyses of this system and the various systems it supported to encompass uncertainty. These margins were specifically identified in a presentation by the licensee's engineers as follows:

- Containment Fan Coolers:

  Their performance was based on 90 percent design air flow, versus 100 percent; 120°F [48.9°C] component cooling water supply temperature, versus the design temperature of 115°F [46.1°C]; and a component cooling water design flow of 1100 gpm [4164 Lpm], versus the flow balance test required minimum flow of 1200 gpm [4542.5 Lpm].

- Shutdown Cooling Heat Exchangers:

  Their performance was based on a 30 percent reduction in the predicted fouled performance; 120°F [48.9°C] component cooling water supply temperature, versus 115°F [46.1°C] design temperature; and a component cooling water design flow of 2550 gpm [9652.8 Lpm], versus the flow balance test required minimum flow of 2800 gpm [10599.2 Lpm].

- Emergency Diesel Generators:

  Heat loads were based on 110 percent generator overload; and design component cooling water flow was 800 gpm [3028.3 Lpm], versus the flow balance required minimum of 850 gpm [3217.6 Lpm].

On the basis of the demonstrated margins, the inspectors estimated that adequate margin existed to account for instrument uncertainties in the testing of the component cooling water pumps.

Chemical and Volume Control Pumps

For these pumps, the licensee's engineers also maintained that, although instrument uncertainty was not explicitly included in the test acceptance criteria or in the supporting analyses, there were ample margins in the analyses to encompass this uncertainty. The worst-case event for these pumps was a small break loss-of-coolant accident, for which 36.6 gpm [138.5 Lpm] was used in the accident analysis. The minimum acceptable inservice testing flow was 40.8 gpm [154.4 Lpm]. This yielded a 4.2 gpm [15.9 Lpm] margin (11.5 percent) between the test value and the analysis value.

The inspectors estimated that sufficient margin existed to account for instrument uncertainties in the testing of the chemical and volume control pumps.

Emergency Feedwater Pumps

For the emergency feedwater pumps, the licensee's representative maintained that there was an ample margin in the design to encompass surveillance test instrument uncertainty. The worst-case accident with regard to these pumps was the main feedwater line break inside containment. This event would rapidly deplete the faulted steam generator water inventory until it reached the feedwater supply ring (38 ft [11.6 m] above the top of the tubesheet). From that point onward the depletion rate would be slower, at the boil-off rate due to the decay heat and sensible heat from the uncontrolled rapid reactor coolant system cooldown. This boil-off would proceed until the steam generator was dry. During this period, no emergency feedwater would be required, since this boil-off would be removing heat at more than the required rate, i.e., a reactor coolant system cooldown. The end of this period would be the first point at which emergency feedwater flow to the intact steam generator would be required. However, at this point, the required heat removal rate, and hence the emergency feedwater makeup demand, would be substantially reduced. This was analytically verified by the licensee's representative while the inspectors were onsite by the performance of an informal analysis using the NRC-approved computer code ABBCE CESEC III.

The formal accident analysis of record, however, differed sharply from this scenario. It contained a very large conservatism, an assumption that the feedwater supply ring was located at the bottom of the generators, just above the tube sheets. The effect of this assumption was to require better emergency feedwater pump performance much earlier in the event. This assumption would cause the entire faulted steam generator water inventory to be very quickly depleted through the break, thereby, not being available for the initial heat removal associated with its boil-off. Additionally, this would also require emergency feedwater along with the intact steam generator to provide heat removal earlier in the event when the decay heat rate would be much higher. Even with this extremely conservative assumption, the analyses showed that the emergency feedwater system could provide the required heat removal with 575 gpm [2176.6 Lpm] (minimum combination of two motor-driven pumps or one turbine-driven pump) delivered to the intact steam generator. This flow represented a 2.3 percent degradation from the motor-driven pump reference curve and a 3.9 percent degradation from the turbine-driven pump reference curve.

Procedure OP-903-046, "Emergency Feedwater Operability Check," Revision 14, Change 5, was the surveillance test procedure used to verify the emergency feedwater pumps' operability. Operability was verified by measuring pump developed head at minimum flow recirculation conditions. The acceptance criteria were developed by projecting back to minimum flow conditions the pump curve that represented the minimum performance limit. This yielded minimum developed head limits of 1268 psid [8742.6 kPad] and 1310 psid [9032.1 kPad] for the motor-driven and turbine-driven pumps, respectively. The surveillance test acceptance criteria represented developed heads of 1278 psid [8811.5 kPad] and 1322 psid [9114.9 kPad], respectively (acceptance criteria converted from psig [kPa] by subtracting 20 psig [137.9 kPa] suction pressure). Therefore, the available margins between the minimum acceptance criteria and the analytical limits were 10 psid [68.9 kPad] (0.8 percent) and 12 psid [82.7 kPad] (0.9 percent), respectively.

Although these margins were potentially inadequate to account for the instrument uncertainty, the inspectors considered the analytical limit margins to be adequate as a result of the very conservative analysis assumption discussed above. Additionally, the inspectors discovered that the test procedure contained requirements to adjust the pump discharge pressure readings for pressure gage inaccuracy (flow corrections would be insignificant and, therefore, unnecessary due to the flatness of the pump curve at minimum flow conditions) before comparing them to the acceptance criteria. Therefore, the inspectors concluded that the instrument uncertainty was adequately accounted for by the procedure.

However, this review revealed a new procedure adequacy concern. Whereas, under design basis conditions, the pumps were required to deliver flow to the steam generators starting at 575 gpm [2176.6 Lpm], this test was being performed at minimum flow conditions. Therefore, these tests were not necessarily representative of the pumps' abilities to perform under design basis conditions. Hence, they could experience degradation at the design basis region of their pump curves that would not necessarily be reflected at test conditions. The licensee's engineers stated that they would consider revising the test procedure to formally include such verifications.

V. Management Meetings

X1 Public Technical Meeting

On February 22, 2000, a technical-level public meeting was held with members of the licensee's staff identified in Attachment 2 of this report. This meeting was held to assist the regional inspectors an opportunity to better understand the licensee's program for addressing instrument uncertainties as applied to safety-related equipment. Attachment 3 to this report contains the handout provided by the licensee at the meeting.

X2 Exit Meeting Summary

The lead inspector presented the inspection results to members of licensee management at the conclusion of the inspection on March 3, 2000. The licensee's representatives acknowledged the findings presented.

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

A followup exit was conducted via teleconference on March 30, 2000, to inform the licensee's staff of a reclassification of certain issues discussed at the onsite exit meeting.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Brandon, Acting Licensing Manager
C. Dugger, Vice President, Operations
E. Ewing, General Manager, Plant Operations
J. Holman, Supervisor, Events Assessments
E. Perkins, Director, Nuclear Safety Assurance
O. Pipkins, Senior Licensing Engineer
R. Putnam, Engineering Supervisor
A. Wrape, Director, Engineering

NRC

T. Farnholtz, Senior Resident Inspector
J. Keeton, Resident Inspector

INSPECTION PROCEDURE USED

IP 92903 Followup - Engineering

ITEMS OPENED AND CLOSED

Opened and Closed

50-382/0001-01	NCV	Failure to assure single failure protection for hydrogen analyzer containment isolation valves (Section E8.1).
50-382/0001-02	NCV	Failure to identify the periodic replacement of elastomeric components in safety-related snubbers (Section E8.6).

Closed

50-382/97032-00	LER	Hydrogen analyzer piping did not meet the requirements for reliable and redundandent containment isolation valves (Section E8.1).
50-382/97032-01	LER	Hydrogen analyzer piping did not meet the requirements for reliable and redundandent containment isolation valves (Section E8.1).
50-382/97033-00	LER	10 CFR 50.46 acceptance criterion exceeded procedure (Section E8.2).

50-382/98017-00	LER	Calculated peak cladding temperature exceeded 10 CFR 50.46 criterion (Section E8.3).
50-382/98022-C	EA	Untimely and inneffective corrective actions (4 examples) (Section E8.4).
50-382/98022-D	EA	Surveillance procedure did not include provisions for assuring that adequate test instrumentation was used (2 examples) (Section E8.5).
50-382/9904-02	URI	Degraded steam generator hydraulic snubbers (Section E8.6).
50-382/9906-04	URI	Application of instrument uncertainties (Section E8.7).

LIST OF DOCUMENTS REVIEWED

Procedures

Design Engineering Administrative Manual No. IC-G-001-02, I & C Uncertainties and Setpoint Calculation Program, dated 8/23/99

OP-903-014, Emergency Feedwater Flow Verification, Revision 10, Change 2

OP-903-046, Emergency Feedwater Pump Operability Check, Revision 14, Change 5

Calculations

EC-195-011, SI-HPSI Flow Instrumentation Loop Uncertainty Calculation, Revision 2

Engineering Request ER-W3-98-0009 Response, dated January 20, 1998

EC-M98-069, HPSI System Performance Surveillance Requirement Basis, Revision 0

EC-C-99-004, Operability Evaluation for Steam Generator Snubbers, Revision 0

Condition Reports

CR-WF3-1997-2695
CR-WF3-1998-1163
CR-WF3-1999-0212
CR-WF3-1999-0213
CR-WF3-2000-0190
CR-WF3-2000-0191
CR-WF3-2000-0192

Miscellaneous

Waterford 3 Technical Specification LCO Grading Results (no document number, no date)

TM-E250.0305, “Vendor Technical Manual for Enertech Hydraulic Shock Suppressors,” Revision 5

Common Cause Analysis for CR 97-2695, dated September 14, 1998

“Request for Quotation for Technical Specification Calculations,” dated October 12, 1998

Enercon letter ENTW-A98-3007, dated October 22, 1998

NRC letter EA 98-022, dated June 16, 1998

NRC letter EA 98-022, dated May 24, 1999

Waterford letter W3F1-98-0119, dated July 29, 1998

Waterford letter W3C3-99-0205, dated June 30, 1999

Engineering Request ER-W3-99-0428, dated April 22, 1999

Engineering Request ER-W3-98-0009

ATTACHMENT 2

LIST OF ATTENDEES AT
PUBLIC MEETING HELD
FEBRUARY 22, 2000,
IN ARLINGTON, TEXAS

Licensee:

J. Holman, Supervisor, Events Assessment
E. Perkins, Director, Nuclear Safety Assurance
R. Putnam, Engineering Supervisor

NRC:

R. Deese, Reactor Inspector, Engineering and Maintenance Branch
C. Paulk, Senior Reactor Inspector, Engineering and Maintenance Branch
D. Powers, Chief, Engineering and Maintenance Branch
D. Prevatte, Consultant, Beckman and Associates

ATTACHMENT 3

LICENSEE HANDOUT
AT PUBLIC MEETING
ON FEBRUARY 22, 2000

NRC AND WATERFORD 3
TECHNICAL WORKING LEVEL MEETING ON
APPLICATION OF INSTRUMENT UNCERTAINTIES
(URI 50-382/9906-04)

1. DESIGN REQUIREMENTS FOR HPSI, ACCW, CCW, CVC, CHW, AND EFW PUMPS

a. Bases for Analytical Value in Design Basis Function

i. High Pressure Safety Injection (HPSI) Pumps

(1) The design basis safety function of the HPSI system is to inject borated water into the RCS to flood and cool the reactor core and to provide for heat removal from the reactor core for extended periods following a LOCA. The HPSI system also has a design basis safety function to inject borated water into the RCS to increase the shutdown margin following a rapid cooldown of the RCS due to a MSLB. The analytical values are derived from the core reload parameters (Safety Analysis Groundrules) and are depicted as an injection flow curve versus RCS pressure. The cold leg injection curve is represented by an assumed injection flow of 777gpm at 0 psia RCS pressure, 371 gpm at 948 psia, and 0 gpm at 1344 psia. The core reload assumptions include a similar injection curve for long term cooling (hot leg injection).

(2) The HPSI pumps are rated for 380 gpm (not including the bypass) at 2830 feet of water (1226.9 psi). Attachment 3 contains the HPSI Pump curves.

ii. Component Cooling Water (CCW) and Auxiliary Component Cooling Water (ACCW) Pumps

(1) The design basis safety function of the CCW and ACCW systems is to remove heat from the containment and reject the heat via the cooling towers to the atmosphere following a LOCA or a MSLB inside the containment. The CCW and ACCW systems also have a design basis safety function to supply component cooling water to the Essential Services Loop (Containment Fan Coolers, Emergency Diesel Generators, Shutdown Cooling Heat Exchangers, and Essential Chillers). The accident analyses assume a heat load of 195.6 Million Btu/hr. CCW is assumed at 4,480 gpm and ACCW is assumed at 5350 gpm.

(2) The CCW temperature control is provided by the dry cooling tower fans or by modulating the ACCW flow through the CCW heat exchanger. CCW temperature control can be provided using both methods, particularly when meteorological conditions are unfavorable. The CCW pumps are rated at 6800 gpm at 145 feet of water (62.9 psi). The ACCW pumps are rated for 6500 gpm at 145 feet of water.

iii. Chemical and Volume Control (CVC) Charging Pumps

(1) The design basis safety function of the CVC is to: 1) maintain the RCS inventory during plant operation, heatups and cooldowns; 2) control boron concentration in the RCS to compensate for reactivity changes and to provide shutdown margin during normal, emergency, maintenance and refueling operations; 3) maintain the chemistry and purity of RCS during startup, normal operation and shutdown; 4) inject concentrated boric acid into the RCS upon a safety injection actuation signal; 5) provide auxiliary pressurizer spray for operator control of RCS pressure during final stages of shutdown and to allow pressurizer cooling during normal operation and during post accident; and 6) provide a controlled path for discharging reactor coolant to the boron management system (BMS) and venting gas to the gas waste management system (GWMS). The Charging pumps are not required for function 6 above. The safety analyses assume that the Charging pumps provide 36.6 gpm for emergency core cooling for small break LOCA events.

(2) The Charging pumps are rated for 44 gpm at 2410 psig.

iv. Essential Chilled Water (CHW) Pumps

(1) The design basis safety function for the CHW system is to supply 42° F chilled water to the air handling unit cooling coils which cool spaces containing equipment for safety related operations during normal and plant accident conditions. The required chilled water flow rate post-LOCA/LOOP is 342.5 gpm. The required chilled water flow rate post-tornado/LOOP is 461.5 gpm.

(2) The chilled water pumps are rated for 510 gpm at 140 feet of water (60.7 psi).

v. Emergency Feed Water (EFW) Pumps

(1) The design basis safety function of the EFW is to provide cooling water to one or both steam generators for the purpose of removal of decay heat from the RCS in response to any event causing low steam generator level coincident with the absence of a low steam generator pressure trip. The safety analyses assume 575 gpm from the turbine driven pump or 575 gpm from both motor driven pumps are available.

(2) The motor driven feedwater pumps are rated for 395 gpm at 1155 psid and are capable of a combined flow rate of 630 gpm at 1102 psig at the entrance of the steam generators. The turbine driven feedwater pump is rated for 780 gpm at 1155 psid and is capable of a flow rate of 645 gpm at 1102 psig at the entrance of the steam generators.

b. Methodology Used to Develop Analytical Value

   i. HPSI Pumps: NRC approved ABB/CE SBLOCA topical methods. Supplement 2 Model (S2M) version of ABB CENP's Small Break LOCA evaluation model as described in CENPD-137P.

   ii. CCW and ACCW Pumps: In order to maximize the heat load on the UHS system, a separate analysis is performed with different assumptions concerning the performance of the Containment Fan Coolers (CFCs) and Shutdown Cooling (SDC) heat exchangers. For this analysis, the CFCs and SDC heat exchangers are clean and receive maximum expected flows. In addition, the CFCs receive 110% of design airflow and the SDC heat exchangers predicted performance is increased by 15%. This conservative approach produces the maximum peak heat on the UHS system of 195.6 Million Btu/hr. The actual expected peak heat load and the minimum required heat load in order to ensure containment integrity is much less. For conservatism in the containment heat removal analysis, the analysis assumes the CFCs and SDC heat exchangers are fouled and receive minimum airflows and cooling water flows. In addition, the predicted fouled performance of the SDC heat exchangers is reduced by approximately 30%. This conservative approach ensures a bounding analysis concerning the heat removal from containment. Overall, the analyses are intentionally conservative and include considerable margin.

   iii. Charging pumps: NRC approved ABB/CE SBLOCA topical methods: Supplement 2 Model (S2M) version of ABB CENP's Small Break LOCA evaluation model as described in CENPD-137P.

   iv. CHW Pumps: The required chilled water flow rate is determined assuming that all of the flow control valves to the individual room coolers are in the full open position except for the Main Control Room and Switchgear areas, which modulate to adjust room temperature. The flowrate in these areas are determined based on maintaining the design temperature using the accident heat load.

   v. EFW Pumps: BTP RSB 5-1

c. Assumptions Used in Determining Nominal Trip Setpoints or Analytical Values

   i. Trip Setpoints

      (1) None of these pumps are used to initiate plant protection action and have no protective action trip setpoints.

      (2) All plant protection system (Reactor Protection System and Engineered Safety Feature Actuation System) setpoints have rigorous setpoint determinations that explicitly consider instrument uncertainties.

ii. Assumptions Used in Determining Analytical Values

(1) HPSI Pump

(a) Appendix K required assumptions

(b) 0.3 Hz under frequency on EDG

(c) Worst case system performance

(2) CCW and ACCW Pumps

(a) CCW is assumed to be at 115° F during the accident.

(b) CCW must provide the cooling capacity to remove the heat generated by the diesel. The diesel is assumed to be loaded at 110% and the coolers are assumed to be at design fouling. The expected diesel loading and fouling during an accident is much less, which would reduce the required cooling requirements.

(c) CCW must provide the cooling capacity to remove adequate heat from the containment atmosphere to ensure the integrity of the containment building. As described earlier, the analysis for determining the minimum heat removal requirements is intentionally conservative and assumes the CFC is at design fouling and 90% airflow. The expected maximum heat removal requirement in an accident is much less than calculated, which would reduce the required cooling requirements.

(d) CCW must provide the cooling capacity to remove heat from the containment sump fluid to ensure the removal of reactor decay heat and the integrity of the containment building. This analysis assumes a worst case conservative decay heat load with the SDC heat exchanger at design fouling. Additionally, for the containment analysis, the heat exchanger performance is reduced by approximately 30%. The expected maximum heat removal requirement in an accident is much less, which would reduce the required cooling requirements.

(e) CCW must provide the cooling capacity to remove adequate heat from the HPSI, LPSI, and CS pump seals to ensure proper operation. This analysis assumes a conservative heat load and cooling requirements were based on the manufacturer's calculations. The expected maximum heat removal requirement in an accident is much less, which would reduce the required cooling requirements.

(f) ACCW must provide the cooling capacity to remove adequate heat from the essential chiller condenser to allow proper cooling of the chilled water system. Chiller load is assumed at a bounding conservative value, the condenser heat exchanger at design fouling and ACCW is assumed at

105° F. The design analyses assume a heat load of 254 tons (A) and 250 tons (B) post-LOCA/LOOP or 233 tons (A) and 232 tons (B) post-tornado/LOOP. The chiller capacity is 325 tons post-LOCA/LOOP and 300 tons post-tornado. The expected cooling flow temperature is 89° F. Thus, the expected maximum heat removal requirement in an accident is much less, which would reduce the required cooling requirements.

(g) ACCW must provide the cooling capacity to remove adequate heat from the CCW heat exchanger to cool CCW to proper temperature for cooling components. The heat exchanger is assumed at design fouling with 5% of the tubes plugged.

(3) Charging Pumps

(a) Appendix K required assumptions

(b) The assumed charging pump flow rate injected into the RCS in the W3 SBLOCA is 18 gpm and is based on a total charging pump flow rate of 36.6 gpm. The 36.6 gpm was derived by allowing the charging pump to degrade to 90% of its rated capacity and subtracting 2% instrument error to establish a worst case flow from one charging pump.

(4) CHW Pump

(a) The calculated pressure loss for the chilled water pump is 61.5 feet at 527.9 gpm. This is substantially below the rated 168 feet for 527.9 gpm. Additionally, the available NPSH is 74 feet. The pump NPSH requirement is only 11 feet. These demonstrate the substantial margin contained in the pump capability.

(5) EFW Pump

(a) 102% power

(b) Maximum CSP temperature

(c) RCPs running and not running

(d) Conservatively assumes quarterly surveillance test is performed with the Condensate Storage Pool full and at the 40° F minimum temperature.

iii. Assumptions Used in Determining Steady State Operating Values

(1) HPSI Pump

(a) Not applicable

(2) CCW and ACCW Pumps

(a) CCW supplies essential and non-essential cooling loops with CCW temperature maintained at about 90° F by DCT or ACCW flow to the CCW heat exchanger. Upon an SIAS, the ACCW pump starts and the CCW temperature control valve setpoint changes to 115° F. CCW temperature will rise and at 102° F, the essential chiller cooling water supply will swap to ACCW. If the CCW temperature rises above 115° F, then the two largest heat loads (CFCs and SDC) will reduce and the heat removal capacity of the DCT increases. Therefore, the UHS response is to mitigate the impact of the increased temperature.

(3) Charging Pump

(a) Anticipated pressurizer volume transients

(4) CHW Pump

(a) The CHW system consists of three 100 percent capacity subsystems with isolation features to assure that chilled water will be provided to the safety related loops following a design basis accident assuming a single failure in any one operating train.

(5) EFW Pump

(a) Not applicable

d. Methods Used to Demonstrate Systems Meet Design Bases Requirements

(1) The testing required by the W3 Technical Specification surveillance requirements assure that the necessary quality of systems and components are maintained, that facility operation will be within the safety limits, and that the limiting conditions for operation will be met.

(2) One of the required tests is W3 Technical Specification 4.0.5. This surveillance requirement requires inservice testing of ASME Code 1, 2, and 3 pumps in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50.55a(g), except where specific written relief has been granted by the Commission. IST for these pumps is performed per ASME Boiler and Pressure Vessel Code Section XI Subsection IWP, which requires testing per OM-6. In general, pump tests are performed by varying the system flow resistance until either the measured differential pressure of the pump or the flow rate equals the corresponding reference value. Test quantities are then measured and compared with the reference value of the same quantity. Any deviations determined are compared with the limits specified in the code and the specified corrective action taken. The procedures that implement the IST for these pumps incorporate the requirements and acceptance limits contained in applicable design documents. The procedures

also require instrumentation to be used that meets the accuracy requirements contained in OM-6. IST of these pumps is typically required quarterly, or at least once per 92 days.

(3) HPSI Pump

(a) Tested per W3 Technical Specifications 4.0.5. In summary, TS 4.05 requires a HPSI pump IST. This is typically implemented concurrently with TS 4.5.2.f.1. Reference flow is 28.9 gpm at an acceptable range of 1429 to 1622 psid (A), 27.6 gpm at an acceptable range of 1429 to 1613.3 psid (B), and 30.5/30.2 gpm at an acceptable range of 1429 to 1597 psid (AB). In all cases, the 1429 is based on the TS 4.5.2.f.1 limit.

(b) Tested per W3 Technical Specification 4.5.2.f.1. In summary, when tested per ASME Section XI code, TS 4.5.2.f.1 requires HPSI differential pressure on minimum recirculation flow to be verified to be greater than 1429 psid. The requirement to verify the minimum pump discharge pressure on recirculation flow ensures that the pump performance curve has not degraded below the level assumed to show that the pump performance supports the safety analysis injection curves.

(c) Tested per W3 Technical Specification 4.5.2.h. In summary, TS 4.5.2.h requires a flow balance test be performed during shutdowns following modifications of the HPSI subsystem that could alter the HPSI subsystem flow characteristics to verify that the 1) sum of the HPSI injection line flow rates, excluding the highest flow rate, is greater than or equal to 675 gpm; and 2) when operating in the hot/cold leg injection mode, the hot leg injection flow must be greater than or equal to 436 gpm and within ±10% of the cold leg flow.

(d) Tested per W3 Technical Specifications 4.5.2.i. In summary, TS 4.5.2.i requires a valve lineup for the system to be performed and the valve response to a SIAS and CSAS to be verified.

(4) CCW and ACCW Pumps

(a) Tested per W3 Technical Specifications 4.0.5. In summary, TS 4.05 requires CCW and ACCW Pump IST. CCW pump reference flow is 4800 gpm at an acceptable range of 65.6 to 80.2 psid (A), 4800 gpm at an acceptable range of 67.8 to 82.8 psid (B), and 4800 gpm at an acceptable range of 68.9 to 84.2 psid (AB). The ACCW pump reference flow is 4050 gpm at an acceptable range of 71.0 to 84.7 psid (A), and 4050 gpm at an acceptable range of 71.0 to 80.6 psid (B). The 71.0 psid is based upon the design limit for the ACCW B pump, as specified by ER-W3-99-0690-00-00.

(b) Tested per W3 Technical Specifications 4.7.3. In summary, TS 4.7.3 requires a valve lineup for the system to be performed and the valve response to a SIAS and CSAS to be verified.

(c) Tested per W3 Technical Specifications 3.7.4. In summary, TS 3.7.4 requires a minimum water level in the WCT basis of 97% and the average WCT basin temperature less than 89° F. These limitations are to ensure that sufficient cooling capacity is available to 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits

(5) Charging Pump

(a) Tested per W3 Technical Specifications 4.0.5. In summary, TS 4.05 requires Charging Pump IST. The charging pump reference discharge pressure is 2350 psig at an acceptable flow rate of 41.7 to 48.3 gpm (A), 2350 psig at an acceptable flow rate of 42.3 to 49.0 gpm (B), and 2350 psig at an acceptable flow rate of 42.0 to 48.6 gpm (AB).

(b) Tested per W3 Technical Specifications 4.1.2.2. In summary, each Charging pump is verified to deliver at least 40 gpm to the RCS every 18 months.

(c) Tested per W3 Technical Specifications 3/4.1.2.4. In summary, at least two Charging pumps are required in modes 1 through 4 and the Charging pump start feature on an SIAS must be verified.

(6) CHW Pump

(a) Tested per W3 Technical Specifications 4.0.5. In summary, TS 4.05 requires CHW Pump IST. CHW pump reference flow is 510 gpm at an acceptable range of 57.2 to 70.0 psid (A), 505 gpm at an acceptable range of 56.6 to 69.2 psid (B), and 510 gpm at an acceptable range of 56.3 to 68.9 psid (AB).

(b) Tested per W3 Technical Specifications 4.7.12.1.b. In summary, TS 4.7.12.1.b requires the chilled water flow to be verified to be ≥ 500 gpm and ≤ 42° F at least every 31 days and flow tested per ASME Section XI every 92 days. The 31 day requirement ensures that the assumptions of the DBA are preserved

(c) Tested per W3 Technical Specifications 4.7.6.3.a. In summary, TS 4.7.6.3.a requires the temperature in the Main Control Room to be verified to be ≤80° F every 12 hours.

(7) EFW Pump

(a) Tested per W3 Technical Specifications 4.0.5. In summary, TS 4.05 requires EFW Pump IST. The EFW pump reference flow is 51.3 gpm at an acceptable range of 1298.0 to 1463.6 psid (A), 47.5 gpm at an acceptable range of 1298.0 to 1442.0 psid (B), and 78.4 gpm at an acceptable range of 1342.0 to 1494.0 psid (AB). In all cases, the low acceptable range is based on the TS limit.

(b) Tested per W3 Technical Specifications TS 4.7.1.2.b. In summary, TS 4.7.1.2 requires each motor driven EFW pump discharge pressure on minimum recirculation flow to be verified greater than or equal to 1298 psig every 92 days on a staggered basis. The turbine driven EFW pump discharge pressure on minimum recirculation flow must be verified greater than or equal to 1342 psig when the S/G pressure is greater than 750 psig every 92 days.

(c) Tested per W3 Technical Specification TS 4.7.1.2.d. In summary, TS 4.7.1.2.d requires, a flow test of at least 700 gpm from the CSP to the S/G's following any cold shutdown of 30 days or longer or whenever feedwater line cleaning through the EFW line has been performed.

e. Methods for Demonstrating Operability of Systems (if different from meeting the design bases)

i. The IST program verifies these pumps perform satisfactory in service. W3 performs IST as required by the W3 Technical Specifications, 10CFR50.55(f), and ASME Section XI to verify these pumps perform satisfactorily in service[1].

ii. Although additional testing beyond that specified by the W3 Technical Specifications is not required to demonstrate adequate inservice performance of these pumps, other test programs may as required by the current licensing bases or by engineering judgement and good practice are performed. Many of these tests, such as system flow balance and heat exchanger performance tests, provide additional assurance that these pumps are capable of meeting their design basis requirements. The methods for demonstrating operability of these pumps include:

(1) HPSI Pump IST, HPSI Full Flow, and HPSI Flow Balance Tests. For a description of these tests see section 1.d.3 above.

(a) HPSI Pump IST is performed every 92 days. The HPSI Pump IST low acceptance criterion is at the TS-LCO limit for the pump. This test verifies the pump differential pressure at minimum recirculation flow.

(b) The HPSI Full Flow Test and Flow Balance Test are performed in refueling outages. These tests verify the pump flow at low RCS pressure conditions.

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[1] The ASME Section XI pump test acceptance criteria considers and is consistent with the applicable design limits as specified by ASME Code, 10CFR50 Appendix B Section XI "Test Control", NUREG-1482.

(2) The ACCW/CCW Pump IST is described in section 1.d.4 above. The ACCW/CCW Flow Balance Test is performed as a repetitive task to verify adequate flow to supplied components. The ACCW/CCW heat exchanger is tested in accordance with GL89-13 as a repetitive task.

(a) The ACCW and CCW Pump IST are performed every 92 days. The ACCW Pump IST low acceptance criterion is at the design basis limit and is based on where the system resistance curve crosses the required pump flow. The CCW Pump IST low acceptance criterion is at the ASME Section XI degradation limit for the pump (90%).

(b) The ACCW/CCW Flow Balance Test is performed each refueling outage.

(c) The ACCW/CCW heat exchanger was last tested in May 1998, and demonstrated 43% of the allowable design fouling including considerations for instrument uncertainties for the test. The SDC heat exchanger was last tested in Feb 1999, and demonstrated 42% (A) and 24% (B) of the allowable design fouling including considerations for instrument uncertainties for the test.

(3) The Charging Pump IST is described in section 1.d.5 above. The Charging Pump Full Flow test is performed to verify charging capability to the RCS.

(a) The Charging Pump IST is performed every 92 days. The Charging Pump IST low acceptance criterion is at the ASME Section XI degradation limit for the pump (95%).

(b) The Charging Pump Full Flow test is performed every 18 months. The low acceptance limit criterion is at the TS limit.

(4) The CHW Pump IST is described in section 1.d.6 above. CHW Pump Flow is periodically verified and a CHW Flow Balance test is performed to verify adequate flow to supplied components. Temperatures in the essential areas cooled by the CHW system is monitored and Main Control Room temperature is required to be verified.

(a) The CHW Pump IST is performed every 92 days. The CHW Pump IST low acceptance criterion is at the ASME Section XI degradation limit for the pump (90%).

(b) The CHW Pump Flow is verified to be greater than 500 gpm every 31 days.

(c) A CHW Flow Balance Test is performed as a repetitive task and verifies required flow is supplied to essential loads.

(d) Temperatures in the essential areas are monitored and temperature in the Main Control Room is required to be verified less than or equal to 80° F every 12 hours.

(5) The EFW Pump IST is described in section 1.d.7 above. EFW Full Flow Testing is performed to verify adequate flow to each steam generator.

(a) The EFW Pump IST is performed every 92 days. The EFW Pump IST low acceptance criterion is at the TS limit.

(b) The EFW Full Flow Test is performed every refueling outage. The EFW Full Flow Test verifies that both motor driven EFW pumps and the turbine driven EFW pump are capable of delivering at least 700 gpm to the steam generators.

iii. In general, the Initial Startup Testing, Inservice Inspection, Surveillance Testing, ASME Section XI tests and other verifications are used to verify that the safety systems are and remain operable. The ability of safety systems to perform their specified functions is verified through a continuous process by surveillances and formal determinations of operability whenever an indication calls into question the ability of the SSC to perform its intended function. This verification is supplemented by ongoing and continuous processes including: day to day operation of the plant, implementation of programs such as IST and ISI, plant walkdowns and tours, observations from the control room, quality assurance activities such as audits and reviews, and engineering design reviews. Once a SSC is established as operable, it is considered reasonable to assume that the SSC should continue to remain operable and that the previously stated verifications should provide that assurance. Other special tests to verify design basis functional capability are not routinely performed unless an indication of potential degradation or nonconforming conditions are indicated. If a degraded or nonconforming condition of these safety related pumps are identified, the Corrective Action Program is utilized to assure timely resolution commensurate with the significance to safety. The impact of any degraded or nonconforming condition on operability is considered within the Corrective Action Program. This approach is consistent with the guidance provided in GL 91-18.

iv. Reasonable assurance of operability is further supported by the diverse and redundant indications available to monitor pump functions. For example Charging pump flow is matched with letdown flow during normal operation to maintain a steady state Pressurizer level. Many instrument indications are available to monitor the overall system function. These redundant and diverse indications were intended by design. It is highly improbable that any combination of instrument uncertainty could prevent the initiation of a safety function or result in a major degradation because: (1) the original design was intentionally conservative, (2) safety instruments are redundant and diverse, (3) drift is periodically detected and corrected, and (4) multiple indications are typically available to monitor system performance. Since all safety channels are redundant, the instruments in both channels would have to drift at the same time in order for major degradation to be undetected, which is highly improbable for independent instruments.

v. In summary, ASME Section XI testing of these pumps is consistent with the applicable design limits and provides a strong assurance that these pumps will perform satisfactory in service. Additional testing programs and other verifications further supplement this assurance of pump operability.

f. Where Uncertainties Addressed

(1) W3 treats instrument uncertainty as described in DEAM procedure IC-G-001-02. This approach was described to NRC management and industry representatives in the Instrument/ECCS Flow Uncertainty Meeting at NRR in Washington, DC on December 2, 1999. NRC management stated that a graded approach was consistent with NRC guidance. Although the NRC did not endorse the EOI approach, NRC management generally supported the bases presented and encouraged W3 participation in industry forums to develop graded approach guidance.

(2) The analysis documented by ER-W3-99-0428-00-00 applied instrument uncertainties to the results of the last IST of the safety related pumps. The engineering document concluded that the application of combined instrument uncertainty to an individual IST of a safety related pump was not required and presented ASME code cases and NRC Guidelines in support of that position. W3 continues to support this position as reasonable and proper. As discussed previously, IST for these pumps is not an individual test. IST for these pumps is required every quarter. The results of the IST for these pumps is compared to a reference baseline and trended. Each test has some level of instrument errors that are random and some that are fixed (bias). The fixed errors, such as tap locations and orifice plate tolerances, are negated by comparing to previous tests. The random errors are also reduced by the comparison to previous tests. With the large number of tests in the IST program for these pumps, the net effect is that the random errors will cancel with time. In addition, for any series of tests, some of these tests will have very small or no random error. The intent of the instrument accuracy requirement in the ASME Section XI code is to manage the random error effects by requiring highly accurate instruments to be used. OM-6a does this by requiring $\leq$2% instrument accuracy (reference accuracy) instruments be used for the test. The net effect is that ASME IST accounts for instrument uncertainty by requiring comparison to a reference baseline and trending pump test data. With hundreds of pump tests performed each year at each plant, the wealth of experience indicates that IST for these pumps is a powerful verification of pump inservice performance.

(3) Pump ISTs performed at minimum recirculation flow result in conservative determinations of pump flow capability due to the shape of the pump curve at minimum recirculation flow conditions. A small flow error that causes the pump to be at a greater than minimum recirculation flow should cause the discharge pressure for the test to be lower than expected. An acceptable differential pressure for the test would conservatively demonstrate the ability of the pump to meet the IST requirement. Conversely, negative flow errors need not be considered since pumps at minimum recirculation flow have zero flow supplied to the system and the pump characteristic is flat at this condition. Thus, flow errors will tend to result in conservative determinations of pump flow capability due to the shape of the pump curve at minimum recirculation flow conditions.

(4) Pump ISTs performed for positive displacement pumps are relatively insensitive to small discharge pressure errors due to the near flat shape of the pump curve. A characteristic of positive displacement pumps is that the flow is insensitive to small changes in discharge pressure.

(5) HPSI Pump

(a) HPSI pump instrument uncertainty is addressed implicitly within the overall margin to safety. The analytical methods used, and the conservative assumptions provided in the analytical methodologies, result in a margin to safety that is much larger than the instrument uncertainties.

(b) The HPSI pump limits specified by 4.5.2.h were screened as 8C, which correspond to grade D. A supplementary analysis, EC-I95-011, was performed in support of the implicit method.

(6) CCW and ACCW Pumps

(a) CCW and ACCW pump instrument uncertainty is addressed implicitly within the overall margin to safety.

(b) The pump flow parameters are not specified in the W3 Technical Specifications and were not screened using the W3 graded approach.

(c) The capacity of the UHS is based on removing the peak heat load during the historical worst combination meteorological condition of 102° F dry bulb and 78° F wet bulb. Peak heat load is about 2 hours in duration. Actual temperatures at the site are typically much more moderate and would provide substantially greater UHS capacity, and cooler ACCW temperatures[2].

(7) Charging Pump

(a) Charging pump instrument uncertainty is addressed implicitly within the overall margin to safety

(b) The Charging pump normally operates during plant operation along with letdown and controlled bleedoff, to maintain a constant pressurizer inventory.

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[2] The NRC staff specifically reviewed the application of flow instrument uncertainty to the CCW and ACCW heat exchanger testing and concluded that the application of instrument uncertainty to these pumps was not required. The apparent violation involving flow instrument uncertainty was withdrawn based upon an "…evaluation that the system was still operable using a rigorous application of instrument uncertainty and based on the lack of explicit regulatory or industry standards/requirements for the application of instrument uncertainties beyond Technical Specification parameters…". This review and conclusion is documented in NRC letter EA98-022 dated June 16, 1998 from Ellis W. Merschoff as item (3) on page 4

Thus, during normal operation this pump is monitored by multiple and diverse indications which mitigate the impact of single instrument drift.

(c) The Charging pump flow limits specified by 4.1.2.2.d were screened as 8C, which correspond to grade D. A supplementary analysis, EC-I95-005, was performed in support of the implicit method.

(8) CHW Pump

(a) CHW pump instrument uncertainty is addressed implicitly within the overall margin to safety.

(b) The CHW pump normally operates during plant operation to supply essential and nonessential loads. Flow through these loads and temperatures of the cooled components are monitored. Thus, during normal operation this pump is monitored by multiple and diverse indications which mitigate the impact of single instrument drift.

(c) CHW pump values are specified in W3 Technical Specifications 4.7.12.1b. The flow and temperature requirements of the chillers ensure safeguards equipment and instrumentation will be cooled during accident conditions. The cooling of this equipment is a secondary or support function. The flow and temperature parameters specified for chilled water are not initial conditions or assumptions in design basis accident analyses. Flow and temperature deviations equivalent to instrument uncertainty would not significantly affect the cooling effect in the safeguards rooms. Chilled water flow and temperature deviations equivalent to instrument uncertainty would not cause prompt failure of safety-related equipment. These parameters were reviewed using the W3 graded approach and were screened as a category 3. Category 3 parameters equate to grade C.

(9) EFW Pump

(a) EFW pump instrument uncertainty is addressed implicitly within the overall margin to safety

(b) EFW pump values are specified in W3 Technical Specifications 4.7.1.2.b. These values are not included in ITS. ITS specifies the test should be performed, but ITS does not include the values. Under ITS, the required test values are controlled under the IST program. These parameters were reviewed using the W3 graded approach and were screened as a category 4. Category 4 parameters equate to grade B.

(10) Implicitly accounting for instrument uncertainty in limits and setpoints for instrument loops which are not safety significant is reasonable based on the large margins to safety and the relatively small instrument uncertainty. While an individual device within an instrument loop might develop an error that could potentially result in some small degradation of safety function, it is highly improbable that a combination of uncertainties could prevent the initiation of a

safety function or result in a major degradation because: (1) safety instruments are redundant and diverse, (2) instrument errors are periodically detected and corrected, (3) all protection system setpoints have been rigorously evaluated to assure safety, and (4) multiple safety trains and systems provide defense in depth for critical safety functions. It is highly unlikely that redundant, diverse, and multiple instruments would drift in the same direction at the same time or for a single pump over multiple tests such that any protracted impact on safety could result.

2. WATERFORD ONGOING GRADED APPROACH TO ESTABLISHMENT OF DESIGN CRITERIA

a. Overview of the Approach

i. This graded approach is intended to implement a graded level of rigor for setpoint and instrument indication limit determination methods based on the safety significance of the instrument function for all values specified in the plant Technical Specifications. Instrument uncertainties for these setpoints and limits are implicitly and explicitly addressed using a combination of qualitative and quantitative methods.

ii. Specific screening criteria was developed by experts from across EOI using the guidelines contained in ISA S67.04 and NRC BTP HICB-12. This criteria is documented in Design Engineering Administration Manual procedure IC-G-001-02.

iii. These screening criteria were applied by an Expert Panel to all of the values specified within the W3 Technical Specifications. The Expert Panel was comprised of a select group of technical personnel having detailed plant, design basis, and safety analysis knowledge and diverse expertise and judgment to determine the safety significance of the instrument function. The Expert Panel was comprised of personnel from the engineering disciplines of instrumentation and control, electrical, mechanical, safety analysis, nuclear and licensing. The team also included an experienced Shift Technical Advisor. The team expertise was supplemented by the results of specific reviews performed by the AE and NSSS vendors to compile the basis for the W3 TS-LCO/SR, the Improved Technical Specification NUREGs, and the on-going efforts to implement the Improved Technical Specifications at W3.

iv. The Expert Panel reviews evaluated each parameter specified within the W3 Technical Specifications. A grading scheme was utilized to determine the safety significance of the instrument function and level of rigor appropriate for the setpoint or instrument indication limit determination methodology. The grade scheme is specific to Waterford 3.

b. Progress for Developing and Implementing

i. The Expert Panel review of the parameters contained within the W3 Technical Specifications is complete. All of the parameters specified within the W3 Technical Specifications requiring rigorous accountings for instrument uncertainty (explicit method) have formal design basis calculations.

ii. Documentation of the W3 Technical Specifications graded approach results is continuing. A draft calculation has been prepared and is about 90% complete. The remaining actions include developing some of the supplemental analyses needed to support some of the grade determinations, independent review, and approval of the new calculation.

iii. We will discuss this draft calculation and status later in the meeting

c. Findings During Implementation of Graded Approach

i. The protection system setpoints have considered instrument uncertainties in a rigorous manner. The protection system setpoints assure with an extremely high probability that protective action will be automatically initiated in the event of an anticipated operational occurrence assuring that specified acceptable fuel design limits will not be exceeded.

ii. The limits specified in the W3 TS-LCO/SR have accounted for instrument uncertainties using a graded level of rigor based on the safety significance of the instrument function. Instrument uncertainties were implicitly and explicitly addressed using a combination of qualitative and quantitative methods.

iii. Non-protection system limits and setpoints generally did not document the manner in which instrument uncertainties was accounted for. However, the reviews demonstrated that the design of the station was conservative and the margins to safety for the instrument setpoint or limit were generally much larger than the instrument uncertainty. No station modifications were performed as a result of these reviews. The conservative margins identified by the reviews supported the determination that the original design considered instrument uncertainty implicitly in the overall margin to safety.

d. Proposed Schedules for Completion

i. The Expert Panel Review is complete.

ii. The formal calculations documenting the rigorous application of instrument uncertainties for those applications having safety significant instrument functions are complete.

iii. The draft calculation that documents the results of the graded approach for the W3 Technical Specification limits is about 90% complete. Completion is anticipated by June 30, 2000.

e. Screening Process Used

i. The screening criteria are provided as Attachment 1. The results of the Expert Panel reviews and application of the screening criteria is provided as a draft calculation.

f. Breakdown of Screening Results

   i. Attachment 2 is a breakdown of the screening results by screening category and applicable grade (safety significance of the instrument function).

g. Parameters Evaluated and Resolved

   i. All parameters contained within the W3 Technical Specifications were evaluated and the level of rigor appropriate to account for instrument uncertainties was resolved (go through an example).

   ii. Operating limits contained in the Emergency Operating Procedure for specifying values at which operator action is needed have accounted for instrument uncertainty as documented by calculation EC-S98-001.

   iii. ASME Section XI testing has accounted for instrument accuracy as specified by OM-6a and documented by engineering document ER-WF3-97-0390-00-00.

3. OTHER APPROACHES TO DEVELOPING SETPOINTS

a. DEAM IC-G-001-02

   i. The W3 approach implements the guidelines contained within DEAM IC-G-001-02. The application and screening criteria is from the DEAM procedure.

   ii. The grading scheme is merely an expression of the same criteria specified in the DEAM procedure in terms of the safety significance of the instrument function. The translation is a direct application of the DEAM guidance and serves as an enhancement for communicating the link between the screening criteria specified in the DEAM and the NRC BTP HICB-12 guidance regarding safety significance of the instrument function.

b. ANSI/ISA S67.04

   i. Does not prescribe any graded approach. The standard committee is drafting a graded approach guideline document, and EOI is participating in that effort.

   ii. ISA S67.04 Part I states that “trip setpoints in nuclear safety-related instruments shall be selected to provide sufficient allowance between the trip setpoint and the safety limit to account for uncertainties. The standard also states that the importance of the various types of safety-related setpoints differ, and as such it may be appropriate to apply different setpoint determination requirements. For automatic setpoints that have a significant importance to safety, for example, those required by the plant safety analyses and directly related to Reactor Protection System, Emergency Core-Cooling Systems, Containment Isolation, and Containment Heat Removal, a stringent setpoint methodology should consider all of the items noted in 4.1 - 4.4.2. However, for setpoints that may not have the same level of stringent requirements, for example, those that are not credited in the safety analyses or that do not have limiting values, the setpoint determination methodology could be less rigorous."

iii. ISA Standard RP67.04 Part II discusses the evolution of the standard and setpoint methodologies and states that this evolution has resulted in the present preference for explicit evaluation of instrument channel uncertainties and resulting setpoints rather than implicitly incorporating such uncertainties into the overall safety analyses. However, the standard goes on to state that both the explicit and implicit approaches can achieve the same objective of assuring that design safety limits will not be exceeded. This position is consistent with the NRC position documented in NRC NUREG-0138.

iv. ISA Standard RP67.04 Part II also states that "During the development of this recommended practice, a level of expectation for setpoint calculations has been identified, which, in the absence of any information on application to less critical setpoints, leads some users to come to expect that all setpoint calculations will contain the same level of rigor and detail. The lack of specific treatment of less critical setpoints has resulted in some potential users expecting the same detailed explicit consideration of all the uncertainty factors described in the recommended practice for all setpoints. It is not the intent of the recommended practice to suggest that the methodology described is applicable to all setpoints. Although it may be used for most setpoint calculations, it is by no means necessary that it may be used for all setpoints. In fact, in some cases, it may not be appropriate…. In applying the standard to the determination of setpoints, a graduated or "graded" approach may be appropriate for setpoints that are not credited in the accident analyses to initiate reactor shutdown or the engineered safety features."

v. ISA RP67.04 Part II also states "it is recognized that some safety-related setpoints are not tied to the safety analyses and do not, even from a system's standpoint, have an explicit limiting value. Thus a graded approach may be applied to the plant's safety-related setpoints. A graded approach might include a method of classifying setpoints according to their contribution to plant safety. Based on the method of classification, the approach would provide guidance on the method to be used to determine the channel uncertainty. Specific criteria for establishing a graded approach or the level of analysis used as part of this type of approach are outside the scope of the recommended practice."

c. Comparison of DEAM Screening Criteria to CE-NPSD-925 Guidelines

i. Similarities

(1) Both approaches use grading schemes intended to be applied to TS-LCO limits and EOP action values

(2) Both approaches intend accounting for instrument uncertainties to be a deliberate effort to determine the impact of instrument uncertainty on safe plant operation

(3) Both approaches are based on a functional review of each instrument application. In this review, the intended function being accomplished by measuring a process parameter was established.

ii. Differences

(1) The CEOG Guideline is general guidance intended for Combustion Engineering plants. The EOI guidance is intended for PWRs and BWRs of different manufacture and design.

(2) The CEOG Guideline does not address instrument uncertainties for instruments where guidance is already available or does not use permanently installed instrumentation. The EOI approach includes these instrument applications.

(3) The CEOG concludes that all safety functions being accomplished by measuring the process parameters that are considered important to safety must have rigorous accountings for instrument uncertainty. That is to say, the CEOG determination of rigor is based on the safety significance of the process parameter being measured. The EOI approach considers the relative magnitudes of the margin to safety and instrument uncertainty to establish the safety significance of the instrument function.

(4) The CEOG classification philosophy and criteria were not provided such that the basis of the level of rigor determinations could be verified. Only generic assessments (high, medium, low) of the safety significance was provided. The EOI approach documents the philosophy, criteria, and basis for the determinations.

(5) The CEOG uses 7 categories:

(a) High degree of safety significance

(b) Moderate to low degree of safety significance

(c) Not safety significant or outside the design basis for plant

(d) Calculated parameters that aren’t process variables

(e) Chemistry values established by laboratory equipment

(f) Instrument application is only used to establish operability

(g) Uses M&TE to determine the parameter value

(6) EOI uses 9 categories (see Attachment 1 for details):

(a) Considers instrument uncertainty for all of the parameters in all of the categories. Instrument uncertainties are considered in establishing all of operational limits and setpoints that control plant operation. These instrument uncertainties may be considered implicitly, explicitly, qualitatively, or quantitatively.

(b) Applied standard criteria to the specific design bases of the plant being evaluated to determine the safety significance of the instrument function.

(c) Applied by personnel having specific and detailed knowledge of the plant design and design bases.