UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 23, 2011

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION TABLE 3.4-1 ISOLATION VALVE ADDITION (TAC NO. ME3421)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 233 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 22, 2010, as supplemented by letters dated June 8 and August 12, 2010, and January 4 and March 7, 2011.

The amendment adds valve SI-4052A (Reactor Coolant Loop (RCL) 2 Shutdown Cooling (SDC) suction inside containment bypass isolation) and valve SI-4052B (RCL 1 SDC suction inside containment bypass isolation) to TS Table 3.4-1, "Reactor Coolant System Pressure Isolation Valves." This bypass line equalizes the SDC system pressure downstream of valve SI-405A (RCL 2 SDC suction inside containment isolation) and valve SI-405B (RCL 1 SDC suction inside containment isolation) in order to minimize the pressure transient in the system when valves SI-405A(B) are opened.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures:
1. Amendment No. 233 to NPF-38
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 233
License No. NPF-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (EOI), dated February 22, 2010, as supplemented by letters dated June 8 and August 12, 2010, and January 4 and March 7, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.2 of Facility Operating License No. NPF-38 is hereby amended to read as follows:

   2. <u>Technical Specifications and Environmental Protection Plan</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 233, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to entering Mode 4 following refueling outage 17.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-38 and
Technical Specifications

Date of Issuance: March 23, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 233

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
3/4 4-20	3/4 4-20

or indirectly any control over (i) the facility, (ii) power or energy produced by the facility, or (iii) the licensees of the facility. Further, any rights acquired under this authorization may be exercised only in compliance with and subject to the requirements and restrictions of this operating license, the Atomic Energy Act of 1954, as amended, and the NRC's regulations. For purposes of this condition, the limitations of 10 CFR 50.81, as now in effect and as they may be subsequently amended, are fully applicable to the equity investors and any successors in interest to the equity investors, as long as the license for the facility remains in effect.

(b) Entergy Louisiana, LLC (or its designee) to notify the NRC in writing prior to any change in (i) the terms or conditions of any lease agreements executed as part of the above authorized financial transactions, (ii) any facility operating agreement involving a licensee that is in effect now or will be in effect in the future, or (iii) the existing property insurance coverages for the facility, that would materially alter the representations and conditions, set forth in the staff's Safety Evaluation enclosed to the NRC letter dated September 18, 1989. In addition, Entergy Louisiana, LLC or its designee is required to notify the NRC of any action by equity investors or successors in interest to Entergy Louisiana, LLC that may have an effect on the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

EOI is authorized to operate the facility at reactor core power levels not in excess of 3716 megawatts thermal (100% power) in accordance with the conditions specified herein.

2. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 233, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

TABLE 3.4-1
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SECTION A	
SI-329A	SIT Check
SI-329B	"
SI-330A	"
SI-330B	"
SI-336A	Cold Leg Injection Check
SI-336B	"
SI-335A	"
SI-335B	"
SI-510A	Hot Leg Injection Check
SI-512A	"
SI-510B	"
SI-512B	"
SI-241	HPSI Check
SI-242	"
SI-243	"
SI-244	"
SECTION B	
SI-142A	LPSI Check
SI-142B	"
SI-143A	"
SI-143B	"
SECTION C POWER-OPERATED VALVES	
SI-401A	SDC Suction Isolation
SI-401B	"
SI-405A	"
SI-405B	"
SI-4052A	SDC Suction Bypass Isolation
SI-4052B	SDC Suction Bypass Isolation

(a) Maximum Allowable Leakage (each valve):

1. SI-4052A(B) leakage limit is less than or equal to 0.375 gpm.

2. Except as noted below, leakage rates greater than 1.0 gpm are unacceptable.

3. For SI-401A(B) and SI-405A(B), leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. For SI-401A(B) and SI-405A(B), leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

5. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 200 psid.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 233 TO

FACILITY OPERATING LICENSE NO. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated February 22, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100550463), as supplemented by letters dated June 8 and August 12, 2010, and January 4 and March 7, 2011 (ADAMS Accession Nos. ML101620462, ML102300177, ML110070026, and ML110670210, respectively), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TSs) for Waterford Steam Electric Station, Unit 3 (Waterford 3). The supplemental letters dated June 8 and August 12, 2010, and January 4 and March 7, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 20, 2010 (75 FR 20633).

The amendment would add valve SI-4052A (Reactor Coolant Loop (RCL) 2 Shutdown Cooling (SDC) suction inside containment bypass isolation) and valve SI-4052B (RCL 1 SDC suction inside containment bypass isolation) to TS Table 3.4-1, "Reactor Coolant System Pressure Isolation Valves." This bypass line equalizes the SDC system pressure downstream of valve SI-405A (RCL 2 SDC suction inside containment isolation) and valve SI-405B (RCL 1 SDC suction inside containment isolation) in order to minimize the pressure transient in the system when valves SI-405A(B) are opened.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The TSs ensure the operational capability of structures, systems, and components that are required to protect the health and safety of the public. The NRC's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," which requires that the TSs include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operations (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

The regulations in 10 CFR 50.36(c)(3) specify that,

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

General Design Criterion (GDC) 17, "Electric power systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 states that:

> An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.
>
> The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.
>
> Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.
>
> Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

GDC 54, "Piping systems penetrating containment," of Appendix A to 10 CFR Part 50, requires that:

> Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus to determine if valve leakage is within acceptable limits.

GDC 55, "Reactor coolant pressure boundary penetrating containment," of Appendix A to 10 CFR Part 50 requires, in part, that:

> Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, 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:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
>
> (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may be not be used as the automatic isolation valve outside containment; or
>
> (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.
>
> Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

The regulations in 10 CFR 50.49, "Environmental qualification of electrical equipment important to safety for nuclear plants," requires an applicant for a license of a nuclear power plant to establish a program for qualifying electrical and instrumentation and controls (I&C) equipment important to safety located in a harsh environment.

The regulations in 10 CFR 50.63, "Loss of all alternating current power," requires, in part, that each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout (SBO).

Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50 relates to safe shutdown requirements.

NRC Regulatory Guide (RG) 1.75, "Criteria for Independence of Electrical Safety Systems," Revision 3, February 2005 (ADAMS Accession No. ML043630448), requires establishing and maintaining the independence of safety-related equipment and circuits, and auxiliary supporting features by physical separation and electrical isolation.

NRC RG 1.63, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants," Revision 3, February 1987 (ADAMS Accession No. ML003740219), requires the external circuit protection of electrical penetration assemblies.

NRC RG 1.155, "Station Blackout," August 1988 (ADAMS Accession No. ML003740034), requires that for the duration of an SBO, the plant be capable of maintaining core cooling and appropriate containment integrity.

NRC RG 1.189, Revision 2, "Fire Protection for Nuclear Power Plants," October 2009 (ADAMS Accession No. ML092580550), provides guidance on the concept of defense-in-depth to fire protection in fire areas important to safety.

NUREG-1432, Revision 3, "Standard Technical Specifications, Combustion Engineering Plants," June 2004, provides the leakage limit guidance.

American Nuclear Society (ANS 56.2)/American National Standards Institute (ANSI) N271-1976, "Containment Isolation Provisions for Fluid Systems," provides minimum design, testing, and maintenance requirements for the isolation of fluid systems which penetrate the primary containment of light water reactors.

3.0 TECHNICAL EVALUATION

3.1 Brief Description of the Amendment Request

The change adds two new 125 Volt direct current (VDC) operated solenoid valves, SI-4052A and SI-4052B, to TS Table 3.4-1, "Reactor Coolant System Pressure Isolation Valves."

Valve SI-4052A, RCL 2 SDC suction inside containment bypass isolation, is installed to bypass SDC isolation valve SI-405A on RCL 2. Similarly, solenoid valve SI-4052B is installed on RCL 1. These lines equalize the SDC system pressure downstream of valves SI-405A(B) in order to minimize the pressure transient in the system when valves SI-405A(B) are opened. These valves are designed to fail-close on loss of power. This provides an additional means of protection against accident or spurious operation. These valves will be normally closed except during SDC operations or if the alignment of Low Temperature Overpressurization relief(s) are required. Existing valves SI-401A(B), which are within the containment and upstream of valves SI-405A(B) and the proposed valves, SI-4052A(B), and valves SI-407A(B), which are outside the containment, are normally closed. The configuration of SI-405A(B) and SI-4052A(B) includes interlocks such that these valves cannot be inadvertently opened with the reactor coolant system (RCS) above the design pressure of the SDC system. The proposed new line will create a bypass around valves SI-405A(B) to minimize the void formed during the plant operating cycle downstream of SI-405A(B) before SI-405A(B) is opened. The proposed solenoid valves SI-4052A(B) will be installed in parallel with valves SI-405A(B) and will be part of the boundary separating Class 1 and Class 2 piping.

3.2 Proposed Changes to TS Table 3.4-1

TS Table 3.4-1, Section C Power Operated Valves

In its letter dated February 22, 2010, the licensee proposed adding the following valves to Section C Power Operated Valves of TS Table 3.4-1:

SI-4052A	SDC Suction Bypass Isolation
SI-4052B	SDC Suction Bypass Isolation

TS Table 3.4-1, Footnote (a)

Currently, the footnote (a) to TS Table 3.4-1 states:

(a) Maximum Allowable Leakage (each valve):

1. Except as noted below, leakage rates greater than 1.0 gpm [gallons per minute] are unacceptable.
2. For power-operated valves (POVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
3. For power-operated valves (POVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are unacceptable.

Revised footnote (a) to TS Table 3.4-1 would state:

(a) Maximum Allowable Leakage (each valve):

1. SI-4052A(B) leakage limit is less than or equal to 0.375 gpm.
2. Except as noted below, leakage rates greater than 1.0 gpm are unacceptable.
3. For SI-401A(B) and SI-405A(B), leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. For SI-401A(B) and SI-405A(B), leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
5. Leakage rates greater than 5.0 gpm are unacceptable.

3.3 NRC Staff Evaluation

3.3.1 Electrical Evaluation

In its application dated February 22, 2010, the licensee stated that the proposed solenoid valves and controls will be powered from the safety-related battery. The licensee evaluated the batteries and power distribution panels (PDPs)and determined that this equipment will have acceptable margin for the proposed modification. The licensee also evaluated the new control circuit lengths for voltage drop concerns and determined that sufficient voltage exists for proper operation. In addition, the licensee stated that the new indicating lights for the proposed valves will be powered from static uninterruptible power supplies (SUPSs) and power distribution panels and determined that these SUPSs and panels have acceptable margins available for the proposed modification.

In its letter dated June 8, 2010, the licensee provided the nameplate information, including ratings, of the proposed valves SI-4052A(B). In addition, the licensee also submitted the electrical single line diagram that shows direct current (DC) control power bus 3A-DC-S and 3B-DC-S and 120 Volt alternating current (VAC) Uninterruptible Power Supply (UPS) 3A-S and 3B-S, and provided schematic drawings that shows the power supplies for valves SI-4025A(B) and the indication lights.

The licensee stated that the SI-4052A(B) solenoid valves and controls will be powered from the safety-related A and B battery, respectively. The indication lights of the valve control are powered from separate safety-related sources UPS 3-A and 3-B, respectively. The licensee's evaluation of the battery loading and the PDPs determined the load on SUPSs 3-A and 3-B are approximately 31 percent and 27 percent of their rating, respectively. Similarly, the loading on the PDP 390-SA and PDP 391-SB are 52 percent and 46 percent of their rating, respectively. Therefore, the addition of the indicating lights for SI-4052A(B) will have an insignificant impact on the SUPS and PDP margin.

The licensee also evaluated the control circuit length for voltage drop with additional loads and determined that adequate voltage is available for proper operation. The lowest design voltages at SI-4052A and SI-4052B were calculated to be 96.9 VDC and 98.7 VDC, respectively, while new solenoid valve has an operating voltage rating range of 90 Volt (V) to 140 V.

The licensee stated that the cables between Remote Shutdown Panel and Auxiliary Relay Panel are Class 1E and are routed in a safety-related raceway that meet all separation, independence, and redundancy requirements and are in accordance with RG 1.75 and the Institute of Electrical and Electronics Engineers (IEEE) Standard 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits." The NRC staff reviewed the modification and concludes that it is consistent with the guidance provided in RG 1.75 and IEEE 384-192; and is, therefore, acceptable.

In its letter dated August 12, 2010, in response to the NRC staff's request for additional information (RAI) dated June 28, 2010 (ADAMS) Accession No. ML110690447), the licensee provided a summary of the protective device coordination analysis for the respective containment penetrations for valves SI-4052A(B). The licensee stated that the electrical penetrations associated with these circuits have double protective devices (two fuses) in each circuit. This is in accordance with Waterford 3's Final Safety Analysis Report (FSAR), paragraphs f and g of Section 8.3.1.1.4, "Electrical Penetrations," and RG 1.63 (October 1973) related to low-voltage circuit protection for containment penetrations. The power sources (power, control, and indication) for the proposed valves SI-4052A(B) are tapped from existing circuits that are currently part of the SRs of TS 3/4.8.4, "Electrical Equipment Protective Devices." Hence, no additional SRs are proposed, which the NRC staff concludes is acceptable.

3.3.1.1 Station Blackout (SBO) Evaluation

In its letter dated January 4, 2011, in response to the NRC staff's RAI dated October 28, 2010 (ADAMS Accession No. ML103010180), the licensee indicated that an evaluation for Waterford 3 was performed in accordance with 10 CFR 50.63, "Loss of all alternating current power," using the guidance in the Nuclear Management and Resources Council, Inc.'s (NUMARC, now Nuclear Energy Institute) NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987 (non-publicly available) and RG 1.155. The licensee indicated that its plant-specific evaluation for Waterford 3 demonstrates that equipment will be functional such that Waterford 3 can safely cope with an SBO for 4 hours. During the 4-hour SBO event, maintaining reactor core cooling for the plant is performed by means of natural circulation. SDC will only be initiated after the 4-hour SBO event. The proposed solenoid valves SI-4052A(B) are required to support SDC only and are not needed for natural circulation after 4-hour SBO event.

Battery capacity calculations verified that the Class 1 E batteries have sufficient capacity to support SBO for 4 hours. Offsite power or the emergency diesel generator is credited for restoration of AC power to the plant ending the 4-hour SBO event. Once AC power is restored, the safety-related battery chargers provide DC power to the DC buses and loads.

The NRC staff concludes that Waterford 3's response for SBO coping is in accordance with 10 CFR 50.63 and consistent with the guidance provided in NUMARC 87-00 and RG 1.155; and is, therefore, acceptable.

3.3.1.2 Evaluation of the Environmental Qualification of the Valves SI-4052A(B) and Associated Components

In its letters dated January 4 and March 7, 2011, the licensee provided details on the environmental qualification (EQ) of valves SI-4052A(B) and the associated components. The licensee also provided a Valcor (valve vendor) similarity report, titled "Similarity Qualification Test Report on Solenoid Valve, Part Number 214167301, Model Number V526-6040-22."

For Waterford 3, valves SI-4052A(B) will be procured to withstand the environmental and accident conditions inside containment as shown in the Waterford 3 FSAR Table 3.11-1, "Environmental Condition Summary." The proposed valves have not been tested for qualification but a similarity report has been submitted by the licensee that describes the similarities and differences between the proposed valve and similar valves which have been previously tested and qualified. This is consistent with 10 CFR 50.49(f)(2), which states,

> Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

The NRC staff reviewed the similarity analysis and the staff's observations are as follows:

The licensee has procured Valcor (the vendor of the proposed valves) valves with Model Number V526-6040-22, for Waterford 3. The licensee stated that these valves are specified as Class 1E, qualified to IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Seismic Category 1, Borated Water Service, EGS (another vendor) type Conduit Seal, Radiation Environment $5.1x10^7$ Rads, and with independent limit switches (2) normally open and (2) normally closed. To qualify this model, Valcor used two comparable valves to evaluate for similarity using the following items:

- Type of technology used in design and manufacture
- Type of critical components
- Mounting/orientation and type of connections
- Service conditions
- Safety functions

The similarities for various type tests for qualifications, such as Design Basis Event environmental and seismic qualification, are extended to the Waterford 3 model because of similarities of material, design, and construction to the tested valves, Model Number V526-6180-1 and Model Number V52600-6042-1A, which is tested for a loss-of-coolant accident (LOCA). The supporting analysis to qualify the proposed valves for Waterford 3 is based upon testing that was performed on the comparable valves and considers the following:

- The materials of construction are similar, considering material properties and aging characteristics.
- The size of the tested and proposed valves is taken into account, considering thermal effects, seismic effects, and the effects of mounting integrity.
- The shape of the tested and proposed valves is evaluated to be the same or similar with any differences considered to not adversely affect the safety functions.
- Operating and environmental stresses on the proposed valves are equal to or less than the tested valves under both normal and abnormal conditions.
- The aging mechanisms that apply to the tested equipment encompass those that apply to the proposed valves.

Both the tested and proposed valves are constructed of the three basic subassemblies, which make up the complete solenoid valve. The tested valves have been tested in accordance with the type tests described in IEEE Standard 323-1974, IEEE Standard 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," and IEEE Standard 382-1985, "IEEE Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants." The body and bonnet of the tested valves have undergone seismic simulation. The solenoid operator of the tested valves has been subjected to thermal aging, radiation aging, seismic simulation, and design basis event (such as LOCA) environment. The plunger and disc assembly of the tested valves has been tested for cyclic aging and seismic simulation.

The similarity report provided highlights of the similarities of the tested and the proposed valves. The environmental qualification of the proposed valves is based on the similarity of construction and materials to the tested valves, with comparison of metallic and non-metallic parts. Some of the highlights are given below.

Both the tested and proposed valves have identical coil-shell assemblies and have identical part numbers (a unique identification number that is considered to be identical in manufacturing process). The differences in the solenoid valves include four switches in the proposed valve to two switches to the tested valves, but the switches are the same part numbers. Based on the above, the NRC staff concludes the coil-shell assemblies and the switches are acceptable.

The metallic materials of construction are not affected by the thermal aging or radiation exposure. However, they are affected by the LOCA chemical spray and saturated steam environments. The similarity report indicated that the tested valves have completed the LOCA test without incurring any corrosion. Based on the above, the NRC staff concludes that the licensee has adequately addressed the environmental qualification of the metallic materials in the tested and proposed valves. The staff concludes that the metallic materials used in the proposed valve and tested valve are the same and, therefore, are acceptable.

The licensee also stated that the non-metallic materials used in the solenoid construction of the proposed valve and of the tested valve are the same and, therefore, the qualification by similarities is extended to the materials in the proposed valve. Based on the above, the NRC staff concludes that this similarity is acceptable.

Structurally, the solenoids on both the proposed and tested valves are of the same three-piece construction, consisting of a coil assembly, and a two-piece solenoid housing, consisting of a side-ported housing and a cover. Both the proposed and tested valves utilize similar O-ring gland configurations. The licensee stated that the solenoid assembly and O-ring seals on the proposed valve are the same materials and method of manufacture as the solenoid assemblies and O-rings qualified for 40 years. The O-ring elastomers were thermally aged to simulate a 10-year minimum qualified life. The tested valve was radiation–aged by exposure to a Cobalt-60 gamma radiation source; the total integrated radiation dose was 9.25×10^7 Rads with no degradation of non-metallics. The required radiation aging for the proposed valve is stated to be 5.1×10^7 Rads. Therefore, the NRC staff concludes that the radiation dose applied to the similar materials in the tested valves bounds the required radiation dose for the proposed valve and is, therefore, acceptable.

The LOCA test is performed to ensure the fully aged solenoid operator is capable of performing its function at the end of its life. Part Number V526-6042-1A was used as the tested valve for LOCA due to the parameters exceeding requirements for the proposed valve and the similarity of construction.

The licensee stated that tested valve was cycled 8,289 times which exceeds the 1-cycle every 18 months' requirement of the proposed valve.

The licensee stated that tested valve underwent successful seismic qualification testing in accordance with IEEE Standard 344-1987. Since the similar tested valve successfully passed seismic qualification and due to the structural similarities between the proposed and tested valves, the NRC staff concludes that the proposed valve also can be considered seismically qualified for the intended application.

The licensee stated that margins are provided in accordance with IEEE Standard 323-1974. The factors that were applied to service conditions for type tests are as follows: Temperature +15 degrees Fahrenheit, Pressure +10 percent, Radiation +10 percent, Voltage +10 percent, double LOCA peak, and Vibration +10 percent.

Other component item differences between the tested valves and the proposed valve are discussed below.

The leadwire connection for the proposed valve is a Quick Disconnect Switch (QDC) which is different from the leadwire connection assembly for tested valves. The QDC is manufactured by same company (EGS) and is qualified by previous tests in accordance with IEEE Standard 323-1974. The QDC is a device that provides a quick electrical disconnect and prevents moisture/water from penetrating into instrument housing, junction boxes. However, the lead-wire switch assemblies are of the same part number. Since the new leadwire assembly had been previously qualified and is functionally identical with identical manufacturing part number, the NRC staff concludes that the QDC is acceptable to be used with the proposed valve.

The licensee stated that the design and technology used to produce Valcor nuclear class solenoid valves is controlled by Valcor's Nuclear Quality Assurance Manual, and that all materials have traceable, auditable links and that the design methods and technology used for all tested and proposed valves are identical. Based on the above, the NRC staff concludes that the licensee has demonstrated similarity with regard to the design and technology used to produce Valcor nuclear class solenoid valves and is, therefore, acceptable.

Additionally, the licensee investigated the environmental qualification of the associated components such as cable, boxes, splices, etc., to be used for the proposed modification. In its letter dated March 7, 2011, the licensee stated, in part, that

> No new cables are installed for this modification. The modification uses existing junction boxes and cables which have been previously qualified... The previously qualified splices are installed in accordance with approved plant procedures and are located above containment flood level which is not subject to submergence. Existing junction boxes are installed with weep holes to prevent moisture from accumulating inside the junction box.

Based on the above, the NRC staff concludes that the licensee has adequately evaluated the impact of the proposed modification on the environmental qualification of the associated components.

Based on the above evaluation, reviewing the similarities and differences, the NRC staff concludes that:

- The proposed valve is similar in material construction, material properties, and aging characteristics to the tested valves.

- The port sizes are different but it has no effect on valve function and, therefore, has negligible impact on similarity.

- The tested and proposed valve sizes are different but it has no impact on the seismic response due to rigidity of the structure and individual seismic and pressure-induced stresses are analyzed and accounted for in the individual design reports.

- The electrical configuration (failure position) of the proposed valve is normally closed versus normally open for the tested valve. The licensee stated that the operability calculation shows that the proposed valve will operate under the worst case condition and therefore flow capacity and failure position of the valve has no effect on valve qualification. Based on this information, the NRC staff concludes that the licensee has demonstrated the environmental qualification of the proposed valve as it related to the electrical configuration (failure position).

- The number of position indication switches are different (4-position for the proposed valve versus 2-position for the tested valve) which is a customer specification. The NRC staff concludes that that the switches are only provided for position indication and have no effect on the valve function and, therefore, there is no impact on proposed valve's environmental qualification.

3.3.2 Containment Evaluation

3.3.2.1 Background

In its letter dated February 22, 2010, the licensee stated, in part, that,:

> The function of the Shutdown Cooling (SDC) System during normal and abnormal operation is to provide a means for removing decay heat from the reactor by providing flow to the reactor core through shutdown cooling heat exchangers. Shutdown Cooling Suction Isolation Valves, SI-405A(B) and SI-4052A(B) are required for containment isolation and they are also the class boundary separating the SDC Class 1 line from the Class 2 low pressure safety injection (LPSI) pump suction piping. SI-405A(B) and SI-4052A(B) are required to be interlocked with pressurizer pressure to prevent over pressurization of the LPSI pump suction piping. The Open Permissive Interlock (OPI) prevents opening the valves until Reactor Coolant System (RCS) pressure is below 386 psia [pounds per square inch, absolute]. Shutdown Cooling entry is not permitted until RCS pressure is less than 392 psia.

3.3.2.2 Containment Analysis

The new line will create a bypass around valves SI-405A(B) to minimize the void formed during the plant operating cycle downstream of SI-405A(B) before SI-405A(B) is opened. The solenoid valves, SI-4052A(B), which will be installed in parallel with valves SI-405A(B), will be part of the class boundary separating Class 1 and Class 2 piping and are identified as containment isolation valves.

The NRC staff reviewed the addition of the two bypass lines for any significant change to the containment net free volume. The staff determined that the proposed change to the containment net free volume is within the documented margin used in the licensee's FSAR containment analysis in comparison to the minimum net free volume. The staff also confirmed that the containment steel volume and surface area limits would be met and that the maximum allowable values would not be challenged. Based on the above, the NRC staff concludes that as a result of the proposed addition of the two bypass lines, the containment integrity would not be challenged since there would be no increase in the maximum containment pressure and temperature.

The NRC staff confirmed that the new bypass line is above the safety injection maximum level. As a result, the staff concludes that no safety-injection sump parameters are challenged and the dose consequences for accidents which are mitigated using the safety-injection sump are not increased.

3.3.2.3 Compliance with GDCs 54 and 55

The proposed new solenoid valve SI-4052A(B) will serve as the inboard containment isolation valve in parallel with existing inboard containment isolation valve SI-405A(B), while existing valve SI-407A(B) will remain as the outboard containment isolation valve. The licensee indicated in its letter dated February 22, 2010, that SI-4052A(B) will be included in Technical Requirements Table 3.6-2, "Containment Isolation Valves," similar to SI-405A(B) to demonstrate compliance with GDC 55.

As described by the licensee in its letter dated February 22, 2010, manual valves SI-4053-A(B) serve no containment isolation function. The open manual valves are in series with solenoid valves SI-4052A(B) and function only to maintain pressure boundary during plant operations. The SI-4053A(B) valves will normally only be closed to facilitate testing when the plant in an outage. Because this open manual valve is never required to perform a containment isolation function, no position indication is required.

GDC 55 states, in part:

> Isolation valves outside of containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

There is no such statement in GDC 55 relevant to isolation valves on the inboard side of containment. Because GDC 55 does not specifically prohibit the location of another valve in the piping between the containment wall and the valves designated as the containment isolation valve, the NRC staff concludes that the GDC requirements are satisfied. In addition, ANSI N271-1976, Figure B-3, "Typical Safety Injection System – High Head Safety Injection Connections," shows a case where an additional valve exists between the designated inboard containment isolation valve and the containment wall.

ANSI N271-1976, Section 5.3.2, "Provisions and Methods," states, in part:

> Provisions shall be made for leakage rate testing of containment isolation valves... The designer must examine the containment isolation provisions for each fluid system that penetrates the containment and determine how each isolation barrier is to be tested ... The designer must then add to the fluid system test and vent connections, test barriers such as additional valves, or other provisions necessary to establish a test volume to conduct the leakage rate tests.

The licensee indicated in its letter dated February 22, 2010, that manual valve SI-4053A(B) is similar in function to the existing test valve SI-4051A(B) as they provide a test barrier, except during normal operation this test barrier valve is left open. The SI-4053A(B) valve is a passive component and functions the same as the piping to maintain a pressure boundary. The NRC staff reviewed this configuration and concludes it is acceptable based on accommodations for testing provisions as specified in Section 5.3.2 of ANSI N271-1976.

The NRC staff concludes that the licensee meets the criteria/requirements of GDC 54 regarding piping systems penetrating containment

3.3.3 Fire Protection

In the approved fire protection program for Waterford 3, the licensee committed to meet the safe shutdown requirements of Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50. In its letter dated February 22, 2010, the licensee stated that the circuit logic and physical design is established to ensure that Appendix R requirements are met. The proposed modification would add only a minimal amount of additional combustible material to the area. The NRC staff concludes that the licensee's statement satisfies the guidance provided by RG 1.189, Revision 2, "Fire Protection for Nuclear Power Plants" and is, therefore, acceptable.

3.4 Conclusion

Based on the above discussions and evaluations, the NRC staff concludes that:

1) The proposed changes to the Waterford 3 TS Table 3.4-1 provides reasonable assurance of the continued availability of the required electrical power to SDC and to maintain the reactor in a safe condition, after an anticipated operational occurrence or a postulated design-basis accident.

2) The proposed changes is in accordance with 10 CFR 50.49, 10 CFR 50.36, 10 CFR 50.63, meets the intent of GDC 17, and the guidance provided in RG 1.63, RG 1.75, and 1.155.

3) The proposed valve is similar or comparable to valves used in similar applications with any differences considered to not adversely affect the safety functions or environmental qualification.

4) The associated components for the proposed modification have been previously environmentally qualified.

5) The proposed changes provide the adequate containment integrity and containment isolation provisions.

6) The proposed changes add only a minimal amount of additional combustible material to the area and, therefore, satisfies the guidance provided in RG 1.189, Revision 2.

7) While the NRC staff's review of the similarity report submitted by the licensee was limited to the review of those areas identified in the above evaluation, the review does not constitute an acceptance of the similarity report.

Based on the above, and the licensee's regulatory commitments described in Section 4.0 below, the NRC staff concludes that the proposed change to TS Table 3.4-1 to add RCS SDC Suction Inside Containment Bypass Isolation valves SI-4052A(B) is acceptable. Furthermore, the staff concludes that the proposed change to the Waterford 3 TSs provides reasonable assurance of the continued availability of the required electrical power to SDC and to maintain the reactor in a safe condition, after an anticipated operational occurrence or a postulated design-basis accident. The staff also concludes that the proposed change is in accordance with 10 CFR 50.49, 10 CFR 50.36, 10 CFR 50.63, meets the intent of GDC 17, and the guidance provided in RG 1.63, RG 1.75, and RG 1.155. Therefore, the staff concludes that the proposed change to TS Table 3.4-1 is acceptable.

In addition, the NRC staff concludes that the licensee has provided adequate justification to support the requested changes and reasonable assurance that Waterford 3 will be able to comply with the regulatory requirements and, therefore, meets 10 CFR 50.36. Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

4.0 REGULATORY COMMITMENTS

In its letter dated February 22, 2010, the licensee made the following regulatory commitments, which are scheduled to be completed prior to Mode 4 following refueling outage 17.

- SI-4052A(B) will be procured as a Safety Class 1 and Seismic Category 1 with a similar code edition as SI-405A(B). The new bypass line will be procured and installed as Safety Related, ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] Section III, Class 1 up to and including solenoid valve SI-4052A(B).

- A code reconciliation will be performed for any components procured to a different code edition than described above.

- SI-4052A(B) seismic qualification will meet the requirements of FSAR Section 3.9C and the new valves will be added to FSAR Table 3.9-9 and 3.9C-1.

- SI-4052A(B) will be procured to withstand the environmental and accident conditions inside containment as shown in FSAR Table 3.11-1.

- SI-4052A(B) will be included in Technical Requirements Table 3.6-2 (Containment Isolation Valves) similar to SI-405A(B) to demonstrate compliance with GDC 55(a).

- Seismic qualifications for the controls will meet IEEE-344-1975 requirements.

The NRC staff considers the above to be regulatory commitments and acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on April 20, 2010 (75 FR 20633). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Swagata Som
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Dan Hoang

Date: March 23, 2011

March 23, 2011

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION TABLE 3.4-1 ISOLATION VALVE ADDITION (TAC NO. ME3421)

Dear Sir or Madam:

The Commission has issued the enclosed Amendment No. 233 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 22, 2010, as supplemented by letters dated June 8 and August 12, 2010, and January 4 and March 7, 2011.

The amendment adds valve SI-4052A (Reactor Coolant Loop (RCL) 2 Shutdown Cooling (SDC) suction inside containment bypass isolation) and valve SI-4052B (RCL 1 SDC suction inside containment bypass isolation) to TS Table 3.4-1, "Reactor Coolant System Pressure Isolation Valves." This bypass line equalizes the SDC system pressure downstream of valve SI-405A (RCL 2 SDC suction inside containment isolation) and valve SI-405B (RCL 1 SDC suction inside containment isolation) in order to minimize the pressure transient in the system when valves SI-405A(B) are opened.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,
/RA/
N. Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382
Enclosures:
1. Amendment No. 233 to NPF-38
2. Safety Evaluation

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