Entergy

Entergy Nuclear South
Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093
Tel 504 739 6379
Fax 504 739 6698
rdodds@entergy.com

R.A. (Al) Dodds, III
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2004-0115

November 19, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Supplement to Amendment Request NPF-38-249
Extended Power Uprate
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy Letter dated November 13, 2003, "License Amendment Request NPF-38-249 Extended Power Uprate"
2. Entergy Letter dated October 18, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
3. Entergy Letter dated May 7, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"
4. Entergy Letter dated July 14, 2004, "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Operating License and Technical Specifications to increase the unit's rated thermal power level from 3441 megawatts thermal (MWt) to 3716 MWt. Section 2.13.6.3.2, of Attachment 5 in Reference 1 provides the analysis results for the steam generator tube rupture event and was supplemented by Reference 2.

On October 14, 2004, Entergy and members of your staff held a call to discuss the assumption that the loss of offsite power (LOOP) would be delayed for 3-seconds following the reactor trip that resulted from a steam generator tube rupture event. As a result of the call, two questions were determined to need formal response. Entergy's response is contained in the Attachment 1.

On November 17, 2004, Entergy and a member of your staff discussed corrections to the reactor coolant system materials list previously provided in Reference 3. The corrections are provided in Attachment 2. The fact that these corrections were necessary has been entered into the 10CFR50 Appendix B corrective action program at Waterford 3

On November 18, 2004, Entergy agreed to provide additional information regarding the turbine overspeed evaluation. This information is provided in Attachment 3.

A001

The no significant hazards consideration included in Reference 4 is not affected by any information contained in the supplemental letter. There are no new commitments contained in this letter.

If you have any questions or require additional information, please contact D. Bryan Miller at 504-739-6692.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 19, 2004.

Sincerely,

R A Dodds, III

RAD/DBM/cbh

Attachments:
1. Response to Request for Additional Information
2. Corrections to Reactor Coolant System Materials List
3. Additional Information Regarding Turbine Overspeed Evaluation

cc: Dr. Bruce S. Mallett
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Senior Resident Inspector
Waterford 3
P.O. Box 822
Killona, LA 70066-0751

U.S. Nuclear Regulatory Commission
Attn: Mr. Nageswaran Kalyanam MS O-7D1
Washington, DC 20555-0001

Wise, Carter, Child & Caraway
Attn: J. Smith
P.O. Box 651
Jackson, MS 39205

Winston & Strawn
Attn: N.S. Reynolds
1400 L Street, NW
Washington, DC 20005-3502

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
P. O. Box 4312
Baton Rouge, LA 70821-4312

American Nuclear Insurers
Attn: Library
Town Center Suite 300S
29th S. Main Street
West Hartford, CT 06107-2445

Attachment 1

To

W3F1-2004-0115

Response to Request for Additional Information

Response to Request For Additional Information Related To Loss of Offsite Power Time Delays Associated With the Steam Generator Tube Rupture Analysis

Question 1:

With regard to the potential Loss of Offsite Power (LOOP) time delays associated with the Steam Generator Tube Rupture (SGTR) at Waterford Steam Electric Station, Unit 3 (Waterford 3), the licensee should provide an evaluation of the Waterford 3 plant-specific design features that justify the use of the chosen time delay for the consequential LOOP. The following possibilities should be addressed for the Waterford 3 site-specific electrical design: degraded switchyard voltage, spurious switchyard breaker-failure-protection-circuit actuation, automatic bus transfer failure, and startup transformer failure. One approach would be to address the events identified in Table G.5 of the "Technical Work To Support Possible Rulemaking For A Risk-Informed Alternative to 10 CFR 50.46/GDC 35" (ML022120661), which indicates that these are the likely causes of a consequential LOOP.

Response 1:

Plant Response to SGTR

For the SGTR event, a reactor trip is initiated due to the Core Protection Calculators (CPC) identifying a hot leg saturation trip. The reactor trip initiates a turbine trip resulting in the turbine valves closing. The main generator reverse power relay detects reverse power and actuates a generator lockout which initiates the opening of the Generator Output Breakers (GOB) and a fast dead bus transfer of the plant auxiliaries.

Background

The safety analysis for the SGTR event credits a 3 second time delay between the reactor trip actuation and Loss of Offsite Power (LOOP). The SGTR time line based on the analyses presented in the Extended Power Uprate (EPU) license amendment request is:

T=0	SGTR occurs
T=~7.5 minutes	Core Protection Calculator hot leg saturation trip
T=~7.5 minutes +3 seconds	LOOP
T=~7.5 minutes + roughly 20 seconds to 1 minute	SIAS on Low Pressurizer Pressure

This 3 second time delay for LOOP for the SGTR event has been implicitly assumed in the Waterford 3 design bases since 1991. The 3 second time delay was credited at that time to alleviate the need for a detailed Departure from Nucleate Boiling Ratio (DNBR) analysis to demonstrate that no fuel failure would occur due to DNBR. Because the SGTR transient documented in the Final Safety Analysis Report (FSAR) was considered more severe, the radiological analysis case assuming the time delay was considered to have DNBR performance which was bounded by the case documented in the FSAR and thus the worst case DNBR transient in the FSAR was not updated. The incorporation into the Waterford 3 licensing and design bases of this 3 second delay was justified based on NRC approval of this assumption for CESSAR and Palo Verde. See Combustion Engineering letter number

LD-82-040, dated March 31, 1982, to Mr. Darrell G. Eisenhut, Docket Number STN-50-470F, for the 3 second bases used in this Waterford 3 analysis.

The 3 second time delay has also recently been approved by the NRC for the Westinghouse AP1000 design.

Waterford 3 Offsite Sources and Grid Analyses

The grid stability studies performed for EPU have demonstrated that the transmission grid remains stable during credible grid transients (See Section 2.3.2 in Attachment 5 of Entergy Operations, Inc. (Entergy's) November 13, 2003 EPU submittal and the response to the Electrical Request for Additional Information (RAI) question 2 in Entergy's April 15, 2004 letter "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"). The response to the Electrical RAI also identified that the Waterford 3 Volt Ampere Reactive (VARs) are administratively limited to minimize the impact of Waterford 3 generation trip on the grid voltage. The Amite South transmission area, where Waterford 3 is located, is fed by 14 transmission lines from east, west and north directions providing diversity. In addition, as indicated in response to TI 2515, Entergy currently performs real time contingency analysis on an N-1 (single generation or transmission element contingency) basis by simulating the loss of transmission facilities at 230kV and above within the Entergy controlled area including Nuclear Power Plants (NPP). This provides post-trip voltages at the NPP with a single unit plant trip only. Entergy Transmission is evaluating enhancements to contingency monitoring software for contingencies greater than N-1 for NPP.

The physical configuration of the Waterford 3 Switchyard (SWYD) and Waterford 3 Switching Station (SWSTA) are designed to minimize degraded switchyard conditions. The Waterford 3 SWYD uses breaker and half scheme with East & West busses. Transmission lines to the Waterford 3 SWSTA from the Waterford 3 SWYD are each tied to the busses by one dedicated breaker (double bus double breaker scheme). All of the breakers in the Waterford 3 SWYD have recently been replaced with SF6 breakers and the two Waterford 3 SWSTA Generator Output Breakers are both being replaced for EPU. In addition redundant batteries are provided for the Switchyard & Switching Station controls. A single failure such as spurious protective relay operation, breaker failure, or transmission line failure associated with the Offsite Power source will be limited to one train of offsite power only.

In view of the Switchyard configuration, equipment upgrades, grid analyses demonstrating ability of the grid system to remain stable after loss of one large generating unit, operating history and EPRI Report 1009110 "The Probability and Consequences of Double Sequencing Nuclear Power Plant Safety Loads R1", a complete LOOP concurrent with a reactor trip at Waterford 3 is not postulated. The following discussion summarizes events related to LOOP concerns due to Waterford 3 plant trip and potential equipment failures.

Waterford 3 Start Up Transformers

Waterford 3 has two normally energized Startup Transformers (SUT), each tied to one of the two transmission lines in the Waterford 3 SWSTA. The non-safety busses (which also feed the safety busses) are normally tied to the Unit Auxiliary Transformers (UAT) and transfer via a fast dead bus transfer to the SUTs on a generator trip.

This configuration provides redundancy whereby, in the remote event that one of the busses fails to transfer, the loads required for plant shutdown will continue to be supplied from the preferred power source on the other bus. Thus, a single failure in the bus transfer scheme will not result in a total LOOP for the plant. A postulated single failure of a breaker to achieve a fast transfer to the SUTs power source would only result in the coast down of two of the four Reactor Coolant Pumps (RCPs).

There are two generator lockout relays used to trip the main generator. Each relay has different input trip signals. A single failure of the lockout relay to trip on reverse power will not result in a LOOP as the generator exciter breaker will not open and the GOBs will remain closed resulting in the plant auxiliaries being fed from the offsite source via the UATs.

The SUTs do not have automatic tap changers. The fixed taps on the transformers have been optimized to maintain adequate bus voltage for varying grid conditions. Tap changer failure is not postulated as there is no tap changer movement required to maintain safety bus voltages.

<u>Waterford 3 Equipment Performance</u>

During normal plant operations, the RCPs are powered from non-Class 1E, 6.9KV AC busses, which are electrically connected through the unit auxiliary transformers and the isolated phase busses to the main generator. Under normal conditions, a fast bus transfer will be initiated upon tripping of the unit auxiliary transformer output breakers, and alternate supply breakers will close within a few cycles to connect the RCP busses to the startup transformers. The startup transformers supply the RCP busses during plant startup or at other times when the main generator or unit auxiliary transformer is out of service.

In the event of a turbine trip during normal plant operations, not involving an electrical fault or other grid disturbances, the main generator will remain synchronized to the high voltage (230KV) grid until residual energy in the turbine is dissipated. The main generator will motor for a short period of time, and will not trip until a sustained reverse power condition exists and the reverse power relay actuates. Reverse power relay actuation initiates the generator lockout relay which will simultaneously trip the generator exciter, the 230 KV GOBs and the unit auxiliary transformer output breakers, thereby initiating a fast bus transfer.

Data from 'Sequence Of Events' (SOE) logs associated with recent plant trips that involved a reactor trip-turbine trip-generator trip from approximately 100% reactor power was reviewed.

On February 13, 2001, Waterford 3 tripped from approximately 80% reactor power during turbine valve testing. The SOE logs indicate that there was a 7 second delay between the reactor trip and the generator lockout relay actuation.

On February 14, 2003, Waterford 3 tripped from approximately 100% reactor power during a bus swap. The SOE logs indicate that there was a 7 second time delay between turbine trip monitored through the 'Auto Stop oil Solenoid' and the generator lockout relay operation.

The above data shows that the time delay between reactor trip or turbine trip and Generator Lockout relay operation is 7 seconds. Therefore, in the event of a turbine trip with the RCP busses connected to the unit auxiliary transformer, the RCPs will receive electrical power for at least three seconds following a reactor trip. A postulated single failure of a breaker to achieve a fast transfer to the start up transformers power source, would result in the coast down of two RCPs only. This maintains reactor core flow high enough so that fuel failure due to DNBR does not occur. Prevention of fuel failure minimizes offsite radiological dose.

If the main turbine were to trip with the RCP busses connected to the startup transformers, the RCPs would continue to receive off-site power without interruption following the turbine trip.

The condensate pump motors and circulating water pump motors have a 6.6 kV voltage rating and are fed from a non-safety related 6.9 kV switchgear powered from the UATs or SUTs. The power to these motors is not available during a LOOP, thus the condenser is not available. For the radiological dose calculations all steam releases immediately after reactor trip are conservatively assumed to go directly to the atmosphere via the atmospheric dump valves or safety valves.

Grid stability analyses performed for EPU indicate that the loss of one large power generating unit on the Entergy South electrical power grid does not lead to grid instability. In the remote event that there are 3 to 4 unplanned contingencies, a loss of a large generating unit may generate voltage deviations in the grid. Under these conditions the resulting electrical system instability may cause a loss of off-site power to that unit. The degree of instability is characterized by the rate of grid voltage degradation, which is dependent upon the magnitude of the load mismatch and the physical parameters of the grid. The physical response of the grid is dependent upon the available spinning reserve and its ability to import power from remaining ties. The grid in the vicinity of Waterford 3 has 14 ties to transmission lines from North, South, East and West. This provides assurance that a plant trip will not lead to unstable voltage conditions. The relatively high number of grid interconnections with the neighboring grid operators provides assurance that the Entergy South grid does not become 'an island' during grid disturbances and hence extended frequency degradation is not anticipated in the vicinity of Waterford 3.

At Entergy South, load shedding is utilized to restore the balance between load and power generation and maintain grid stability. The limiting criterion for power transfer is the thermal limit of transmission lines. This thermal limit is reached prior to when degraded voltage conditions are experienced on the grid in the vicinity of Waterford 3. Grid operators are sensitive to power transmission limitations during peak summer periods and take corrective actions when the transmission system is approaching its thermal limits.

When load shedding or other corrective actions are not sufficient to avert voltage degradation, loss of off-site power to the plant can occur as a result of that plant tripping offline. For Waterford 3, a degraded voltage of 93.1% at the 4160V safety bus would result in a LOOP event.

If the generation loss, due to the plant trip, were to cause a degraded voltage on the safety busses, the degraded voltage relays have a 12.5 second time delay to trip the busses. This time is well beyond the 3 second assumption in the SGTR event.

Waterford-3 Technical Specification 3.1.3.4 requires that the average CEA drop time from a fully withdrawn position until the CEA reaches the 90% insertion position to be less than or equal to 3.0 seconds. Because of this, DNBR performance is not in question with the assumption of a 3 second time delay between reactor trip and LOOP. RCPs would continue to run and provide flow for this time period, thus providing adequate core cooling. The primary concern for DNBR performance for SGTR is not the slow depressurization of the RCS associated with the tube rupture, but rather the sudden decrease in RCS flow if a LOOP occurred while the reactor was still at 100% power conditions. Similarly, while safety injection flow is credited in the response of the Nuclear Steam Supply System (NSSS) subsequent to the tube rupture, the analyses are relatively insensitive to the timing of safety injection flow initiation. As documented in Table 2.13.6.3.2-2 provided in Entergy letter W3F1-2004-0096 dated October 18, 2004, a Safety Injection Actuation Signal (SIAS) is predicted to occur some 40 seconds following reactor trip. The assumed safety injection flow is biased toward high flow to maximize RCS pressure, and thus result in increased releases of primary coolant activity from the RCS to the steam generator. Thus, for purposes of the radiological analysis, a delay in delivery of safety injection flow could result in a small decrease in releases to the secondary side and thus in radiological dose associated with the event. Because the releases are dominated by the operator actions to cooldown the plant to shutdown cooling entry conditions over a postulated 8 hour time period, there is very little impact due to delays on the order of the 10 second diesel start time on the delivery of safety injection flow.

Question 2:

Because the SGTR event involves the actuation of Emergency Core Cooling Systems (ECCS) systems, the consequences of the delayed LOOP on the performance of the electrical ECCS systems should be evaluated. The consequences of double sequencing and its associated vulnerabilities that would occur as the result of the delayed LOOP should be a part of this evaluation. These vulnerabilities include, but are not necessarily limited to: the consequences of starting large continuous-duty motors twice in quick succession with the first start under degraded voltage conditions and the second start with pump discharge valves open; the adequacy of the existing control logic to start loads on offsite power, shed those loads following the LOOP, and subsequently re-sequence those loads on the EDG with necessary delay to allow motor residual voltage to decay; interaction between the double sequencing and circuit breaker anti-pump logic that could lock out the breakers; the capability of the safety batteries to operate the necessary systems during an initial offsite power degraded voltage ECCS start and subsequently restart the ECCS on Emergency Diesel Generators (EDGs); and the potential to trip motor overload protection or blow fuses as a result of a degraded voltage double sequencing scenario. Electric Power Research Institute (EPRI) technical report 1009110, "The Probability and Consequences of Double Sequencing Nuclear Power Plant Safety Loads, Revision 1," dated October 2003, can be used as a resource for the electrical issues. The staff has not approved this report, therefore the report must be used together with the staff electrical comments on the report found in Adams Accession No. ML042600254 on the NRC Adams Document Control System.

Response 2:

As stated in the response to Question 1, the SGTR time line is:

T=0	SGTR occurs
T=~7.5 minutes	Core Protection Calculator hot leg saturation trip
T=~7.5 minutes +3 seconds	LOOP
T=~7.5 minutes + roughly 20 seconds to 1 minute	SIAS on Low Pressurizer Pressure

Assuming that the SGTR sequence of events results in a reactor trip and a consequential LOOP after 3 seconds, it is postulated that SIAS would occur approximately 20 seconds to 1 minute after the LOOP. Thus the LOOP would occur prior to SIAS which would be expected roughly 20 seconds to 1 minute after the CPC initiated reactor trip.

Assuming that there is a LOOP 3 seconds after the reactor trip then the Emergency Diesel Generator (EDG) is started and loaded normally (see FSAR Figure 8.3-27) without double sequencing since the SIAS has not yet occurred. When the SIAS occurs, the sequencer is reset with a 2 second time delay and additional ECCS loads are sequenced on as necessary. In the remote event that SIAS is concurrent with a large load (e.g. chiller) starting during the LOOP initiated EDG start, there will be a 2 second delay for the sequencer to reset and 1.5 second delay before the first large High Pressure Safety Injection (HPSI) motor starts. There is no potential for simultaneous start of large motors. Waterford 3 EDGs are capable of starting and accelerating two large motors simultaneously with voltage remaining above 75% of nominal as documented in the EDG dynamic loading analyses.

If a LOOP occurs after the SIAS, then there is a potential for double sequencing the ECCS loads. The most limiting case of double sequencing would occur after a CPC generated plant trip followed by SIAS, expected to occur roughly 20 seconds to 1 minute later. The SIAS would result in starting the EDGs. The EDG output breaker would remain open and the ECCS loads would be sequenced onto the busses powered from the offsite source. If the grid was stressed and the additional auxiliary plant load results in degraded voltage conditions such that the 4160V safety bus degraded voltage relays are actuated during the sequencing, then there is a potential to re-sequence the required ECCS loads. The degraded voltage relays are set to actuate at 93.1% bus voltage and have a 12.5 second time delay.

For the SGTR event, the load of interest is the high pressure safety injection pump. This is the largest rotating component that is sequenced in the 1.5 second load block after an SIAS without LOOP. This motor will have minimum cooldown time during double sequencing. Subsequent motor loads are smaller and have longer time interval between successive starts.

Consider the case where the ECCS load sequencing is initiated by an SIAS with offsite power available and the turbine and generator tripped. The EDGs will get an emergency start but will run in standby mode with output breakers open. The sequencing of events is as follows:

T=0	SIAS occurs. EDG starts in emergency mode and runs with output breaker open. Normal plant auxiliaries continue to function as designed for emergency operation.
T=2.0 seconds	Sequencer resets.
T=3.5 seconds	HPSI pump starts and Degraded Voltage relays actuate due to safety bus voltage degrading to approximately 90% of 4160V.
T=~ 6.5 seconds	Spring Charging of the HPSI breaker is complete and anti pump

	relay de-energized.
T=9.0 seconds	Shield building ventilation system, Controlled ventilation system and Containment Fan coolers (if not running) start.
T=9.5 seconds	HPSI pump at full speed under reduced voltage conditions.
T=16 seconds	Degraded Voltage relays trip Safety related busses and isolate non essential loads. Plant safety busses isolate from grid. Sequencer resets.
T= 21 seconds	EDG output breaker closes. Safety bus undervoltage relays reset and sequencer starts. Undervoltage override relay actuated to protect sequencer resets during initial few loading blocks.
T= 22.5 seconds	HPSI pump breaker closes.
T= 25.1 seconds	HPSI pump at full speed at full voltage.
T= 25.5 seconds	Spring Charging of the HPSI breaker is complete and anti pump relay de-energized.
T= 28 seconds	The '7 second' loads are sequenced. This load block includes battery chargers.

Note: The timeline above identifies only the loads of interest for the discussion. Other equipment such as transformers, valves, dampers etc. that are energized / de-energized during this period will function as designed.

The HPSI motor is expected to function satisfactorily during this transient. This is based on:

- The HPSI pump motor is designed to start at 75% voltage and accelerate to full speed in 6 seconds at the reduced voltage. Hence operation at degraded voltage (approximately 90% of nominal) conditions for 12.5 seconds is not expected to excessively overheat the motor. The off-site voltage remains above 0.97 Per Unit (PU) under various transmission contingencies while Waterford 3 is off-line (see response to the Electrical Request for Additional Information (RAI) question 2 in Entergy's April 15, 2004 letter "Supplement to Amendment Request NPF-38-249 Extended Power Uprate"). At 97% grid voltage the 4.16 kV bus voltage level is above the 90% degraded voltage level as concluded in the degraded voltage calculation.
- The HPSI motors are designed to be capable of six consecutive starts with the motor initially at ambient temperature per the motor nameplate. Hence two rapid starts (one at 90% degraded voltage) will not damage the motor.
- The discharge lines are expected to remain filled with water. The potential for water hammer is minimal.
- The HPSI pump motors are designed to start the pumps against check valves, normally open motor operated discharge valve, and partially open flow control valves, all in series. The stroke time for the HPSI flow control valves is 10 seconds. The motor operated flow control valves start to open immediately upon SIAS. During an SIAS event without LOOP, the HPSI motor is sequenced on the safety bus 3.5 seconds after the SIAS at which time the HPSI flow control valves are at approximately 35% opening. Thus the issue of starting HPSI motors with the outlet valves in the open position is within the normal design configuration.

- Simultaneous start of two large motors from different load blocks is not postulated due to the existing design of the sequencer.
- There is adequate time (6.5 seconds) between stopping from rated speed to restarting such that there are no residual voltages leading to out of phase closure/restart of motor.
- The timing of the sequencer prevents interaction between the double sequencing and circuit breaker anti-pump logic that could lock out the breakers.
- HPSI motors are protected by overcurrent relays. There are no solid state protective relays with thermal memory capability installed in the plant for protection of motors. Hence motor tripping from these types of relays, during a double sequencing event, is not postulated. The motor starting and operation at approximately 85% bus voltage (loads sequencing on the EDGs) coupled with a 6.5 second dead time is not expected to lead to inadvertent overload relay operation. The simulated generator voltage profile illustrates approximately 85% bus voltage (momentary) and recovers during each load block based on sequencer loading as documented in the EDG dynamic loading analysis. Fuses and thermal overloads associated with smaller motors are not expected to trip due to the time interval between successive starts.
- High Pressure Safety Injection Pump Motor maintenance is performed every 36 months under Electrical Maintenance procedure ME-004-345, "High Pressure Safety Injection Pump Motor". Maintenance includes the following; perform resistance-to-ground measurements using a megohmmeter (M&TE) and phase-to-phase resistance values (M&TE), motor is checked to ensure that it is dry and free of loose dust, dirt, rust and corrosion buildup, remove the main junction box cover and verify that the connections and tape are not damaged or heat degraded, and check the resistance of the grounding cable at the motor's casing.

  On-line and Off-line motor testing is performed under Electrical Maintenance procedure ME-007-057, "MCE/EMAX Data Acquisition" for trending purposes only. This testing includes a Standard AC test (phase-to-phase resistance imbalances, phase-to-phase inductance imbalances, capacitance-to-ground measurements), Polarization Index test, and a Step Voltage test.

  Vibration readings are taken under DC-310, "Predictive Maintenance Program" for trending purposes on the motor during the quarterly HPSI Pump In-Service Test (IST). Thermography scans are also taken periodically during pump ISTs. High Pressure Safety Injection Pump motor bearing oil samples are taken every 9 months and analyzed for particulates, viscosity, water, and bearing wear metals.

The normal running ECCS loads such as the Containment Spray Pumps, Component Cooling Water pumps, Auxiliary Component Cooling Water pumps, chillers and HVAC fans are specified to start and accelerate driven equipment at 90 percent rated voltage at rated frequency without exceeding the permissible temperature in accordance with National Electric Manufacturers Association (NEMA) Standards MG-1. NEMA Standards MG-1 specifies "two starts in succession, coasting to rest between starts, with the motor initially at ambient temperature." The original Low Pressure Safety Injection (LPSI) motors are capable of six consecutive starts with motor initially at ambient temperature. One of the LPSI motors was replaced with a Westinghouse motor and is capable of two consecutive starts with the motor initially at ambient temperature. Therefore, these motors are expected to successfully sequence on to the EDGs. These loads are smaller and have longer time intervals between successive starts. Some of these loads are tripped and sequenced onto the EDG per surveillance testing every 18 months.

Waterford 3 motors are specified to the typical motor rated voltage (460V, 4000V, and 6600V). The safety related ECCS motors such as the HPSI, LPSI, containment spray pumps, component cooling water pumps, Auxiliary component cooling water pumps, chillers and HVAC fans have a service factor of either 1.0 or 1.15. Similarly, the typical smaller motors (460 volt and 250 HP and below) have a service factor of either 1.0 or 1.15. The thermal overload sizing is selected based on the NEMA size starter and the full load motor current. The thermal overload for those motors with a service factor of 1.15 is selected one size larger than the motors with the same Full Load Ampere (FLA) at a service factor of 1.0.

The thermal overload relays for safety related Motor Operated Valves (MOVs) are bypassed to prevent tripping when an Engineered Safety Features Actuation Signal is present. The safety related 480V Motor Control Centers (MCCs) and all associated control devices (circuit breakers, in-line fuses, motor starters, heater overload elements, control fuses) were furnished as one assembly. The motor protective devices were designed to provide overcurrent and locked rotor protection when the motor operates within the range of 110 to 90 percent of its rated voltage and under the abnormal electrical supply condition (336 volts recovers to 408 volts in 2 seconds). Therefore, it is not postulated that the thermal overload devices, in-line fuse, and control fuses (protecting the contact coils) will interrupt as a result of the degraded voltage double sequencing condition.

Each EDG has adequate capacity to start and accelerate the largest single load out of sequence, with all other loads running. This provides additional assurance that double sequenced loads can be started on an EDG even if there is minor overlap in the loading sequence.

The safety related batteries are sized for peak DC loads for 1 minute. For the SGTR event discussed above (SIAS followed by LOOP), the batteries carry DC loads for less than 1 minute, including ramp up time for the battery chargers. The ramp up time is expected to be less than 20 seconds based on conversation with C&D Technologies, Inc., manufacturer of Waterford 3 battery chargers. The DC system is adequately sized for the potential double sequencing operations described above.

Attachment 2

To

W3F1-2004-0115

Corrections to Reactor Coolant System Materials List

Corrections to Reactor Coolant System Materials List

In Attachment 1, Response 1 of the Entergy Operations, Inc. (Entergy) May 7, 2004 letter information was provided on reactor coolant system (RCS) materials. Corrections to information in Tables 2 and 3 of this Response 1 are identified below along with a discussion of the significance of each correction. The corrections identified below are the result of review of additional documentation and references for the materials of construction. None of the corrections are considered to be technically significant with respect to the integrity of the reactor coolant system pressure boundary. Revised Tables 2 and 3 from the May 7, 2004 response showing the corrections are also provided below

Table 2	Explanation of Correction
Pressurizer	
Heater Sleeve Plug - Material Specification is SB-167	This correction only changes the starting product form of the material from bar to tubing. The grade of material, mechanical and corrosion properties are the same for both product forms.
Studs and Nuts - Material Specification should include SA-193-B7	Correction adds an additional type of low alloy steel bolting material that is commonly used in primary closure applications. The same two material designations are identified for the steam generator studs and nuts.
Steam Generator	
Secondary Shell and Head - Material Specification should include SA-533 Gr. B Class 1	SA-533 Gr. B Class 1 plate was identified as an additional material of construction. This plate material has the same mechanical property requirements as SA-533 Gr. A Class 1, the material originally listed.
Valves	
Bonnet - Material Specification should remove SA-276 F316	This particular bar product form specification was not identified as being used for the valve construction. Another bar product form specification, SA-479, for Type 316 stainless steel is still included in the list of construction materials.
Disc or Poppet - Material Specification should remove SA-276 F316	This particular bar product form specification was not identified as being used for the valve construction. Another bar product form specification, SA-479, for Type 316 stainless steel is still included in the list of construction materials.

Table 3	Explanation of Correction
No. 5 - should be removed from the listing; there are no locations where SA-508 Class 1 is welded to SA-508 Class 1.	This item was removed because no weld seam locations were identified for this combination of base materials. The weld filler metal previously listed for this combination is still included in the table for other base metal combinations, where it is applicable.
No. 7 - Material Specification should list SFA 5.11, E-NiCrFe-3	This change corrects the specification number and designation for shielded metal arc electrodes for this filler metal that was incorrectly identified by the designation for the bare wire form of this filler metal. The mechanical properties are the same for both filler metals.
No. 9 - Material Specification should list SFA 5.11, E-NiCrFe-3	This change corrects the specification number and designation for shielded metal arc electrodes for this filler metal that was incorrectly identified by the designation for the bare wire form of this filler metal. The mechanical properties are the same for both filler metals.
No. 12 - Material Specification should list SFA 5.11, E-NiCrFe-3	This change removes the carbon and low alloy steel filler metals incorrectly identified for this bi-metallic weld combination of base metals and adds the correct specification number and designation for shielded metal arc electrodes for this filler metal. This same filler metal is listed in Table 3 for other locations of RCS welds for other combinations of base materials.
No. 15 - 2nd 'Base Material' should list SB-166	This correction only changes the starting product form of the material from tubing to bar. The grade of material, mechanical and corrosion properties are the same for both product forms.
No. 16 - Material Specification should list a. SFA 5.1, E-7018 b. MIL-E-18193-B4	This correction changes the shielded metal arc welding electrode from an 80 ksi tensile strength (E-8018, C3) to a 70 ksi tensile strength (E-7018) electrode and adds a wire specification for welding of this combination of filler metals. These same filler metals are listed in Table 3 for other locations for RCS welds in other combinations of these base materials with other carbon and low alloy steel materials.

Table 2
REACTOR COOLANT SYSTEM MATERIALS
(Sheet 1 of 5)

Component	Material Specification
Reactor vessel	
Shell	SA-533 Grade B Class 1
Forgings (flanges, nozzles, and safe ends)	SA-508 Class 1 or Class 2
Cladding [(a)]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite or NiCrFe alloy
Reactor vessel head CEDM Nozzles [(a)]	SB-166
Instrument nozzles [(a)]	SB-167 and SA-182, F-304
Control element drive mechanism housings	
Lower [(a)]	SA-182 Type 403 stainless steel Special Code Case 1334 with end fittings of SB-166
Upper [(a)]	SA-213 Type 316 stainless steel with lower end fitting of SB-166 and upper end fitting of SA-479 Type 316, vent valve seal of ASTM A276 Type 440C stainless steel seat
Closure head bolts	SA-540 Grade B24
Pressurizer	
Shell [(a)]	SA-533 Grade B Class 1 (A gap exists between the original Inconel 600 and replacement Inconel 690 materials on the repaired instrument nozzles and heater sleeves.)
Shell Cladding [(a)]	Weld deposited austenitic stainless steel with greater than 5 percent delta ferrite or NiCrFe alloy
Forged nozzles	SA-508 Class 2
Instrument nozzles [(a)]	SB-166
Surge and safety valve nozzle safe ends	SA-351 Grade CF8M

Table 2
REACTOR COOLANT SYSTEM MATERIALS
(Sheet 2 of 5)

Component	Material Specification
Heater sleeves	SB-167
Heater sleeve plug	~~SB-166~~ SB-167
Studs and nuts	~~SA-540 Grade B24~~ SA-193-B7
Steam Generator	
Primary head	SA-533 Grade B Class 1
Primary nozzles and safe ends	SA-508 Class 1 and Class 2
Primary head cladding [a]	Weld deposited austenitic stainless steel with greater than 5 percent delta ferrite
Tubesheet	SA-508 Class 2
Tubesheet stay	SA-508 Class 2
Tubesheet cladding [a]	Weld deposited NiCrFe alloy
Tubes [a]	NiCrFe alloy (SB-163)
Secondary shell and head	SA-533 Grade A Class 1 ~~SA-516 Grade 70~~ SA-533 Grade B Class 1
Secondary nozzles	SA-508 Class 1 and Class 2
Secondary nozzle safe ends	SA-508 Class 1
Secondary instrument nozzles	SA-106 Grade B
Studs and nuts	SA-540 Grade B24 and SA-193 Grade B7
Reactor Coolant Pumps	
Casing [a]	SA-351 Grade CF8M
Pump cover (lower flange of driver mount)	SA-105
Cladding [a]	Austenitic steel wire electrodes conforming to requirements of ASME/AWS SFA/A-5.4 and SFA/A-5.9 Type 308 or 309

Table 2
REACTOR COOLANT SYSTEM MATERIALS
(Sheet 3 of 5)

Component	Material Specification
Bolts	SA-540 Gr B23 Class 4 SA-564, Type 630, H-1100 (For seal cartridge and seal heat exchanger)
Nuts	SA-194 Grade 7 SA-564, Type 630, H-1100 (For seal cartridge and seal heat exchanger)
Heat exchanger flange	SA-240 TP 304 Annealed or SA-182 Grade F304
Reactor Coolant Piping	
Piping (30" and 42")	SA-516 Grade 70 (SA-264 Clad Plate) [b]
Cladding [a]	SA-240, 304L
Surge line (12") [a]	SA-351 Grade CF8M
Piping [a]	
Pressurizer spray	SA-376, TP-304
Shutdown cooling return	SA-376, TP-304
Reactor coolant drain	SA-376, TP-316 or TP-304
Charging line	SA-376, TP-304
Safety injection	SA-376, TP-304
Letdown line	SA-376, TP-316 or TP-304
Shutdown cooling bypass	SA-376, TP-304
Piping Nozzles and Safe Ends [a]	
Piping safe ends (30")	SA-351 Grade CF8M
Surge nozzle forging	SA-105 Grade II
Surge nozzle safe end	SA-351 Grade CF8M
Shutdown cooling outlet nozzle forgings	SA-105 Grade II

Table 2
REACTOR COOLANT SYSTEM MATERIALS
(Sheet 4 of 5)

Component	Material Specification
Shutdown cooling outlet nozzle safe ends	SA-351 Grade CF8M
Safety injection nozzle forgings	SA-182 F1
Safety injection nozzle safe ends	SA-351 Grade CF8M
Charging inlet nozzle forging	SA-182 F1
Charging inlet nozzle safe end	SA-182 F316
Spray nozzle forgings	SA-105 Grade II
Spray nozzle safe ends	SA-182 F316
Letdown and drain or drain nozzle forgings	SA-105 Grade II
Letdown and drain or drain nozzle safe ends	SA-182 F316
Sampling or pressure measurement nozzles	SB-166
Sampling or pressure measurement nozzle safe ends	SA-182 F316
RTD nozzles	SB-166 and SA-182 F316
Sampling nozzle (surge line)	SA-182 F316
Valves[a]	
Body	SA-182 F316, SA-479 Type 316 and SA-351 Grade CF8M
Bonnet	SA-105 Grade II, SA-351 Grade CF8M, SA-479 Type 316, ~~SA-276 F316,~~ SA-240 Type 316 and SA-182 F316
Disc or Poppet	SA-637 Grade 688, SA-240 Type 316, ~~SA-276 F316,~~ SA-479 Type 316, SA-182 F316, SA-351 Grade CF8M, SA-351 Grade CF3 and SA-564 Grade 630

Table 2
REACTOR COOLANT SYSTEM MATERIALS
(Sheet 5 of 5)

Component	Material Specification
Mechanical Nozzle Seal Assembly (MNSA)	
Assembly	SA-479 Type 304
Seal[(a)]	Grafoil Grade GTJ Nuclear Grade A
Bolting Material	SA-453 Grade 660

[(a)] – Materials exposed to reactor coolant

[(b)] – Material within 4 inches of weld centerline on Field Welds P1OW1 and P1OW2 have been rated with a strength level of 65 ksi per CE Analytical Evaluation Report CENC-1460.

Table 3
WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS
(Sheet 1 of 2)

	Material Specification	Base Material		Weld Material
1.	SA-533 Grade B Class 1	SA-533 Grade B Class 1	a. b.	SFA 5.5, (a) E-8018, C3 MIL-E-18193, B-4
2.	SA-508 Class 2	SA-533 Grade B Class 1	a. b.	SFA 5.5, E-8018, C3 MIL-E-18193, B-4
3.	SA-508 Class 1	SA-508 Class 2		SFA 5.5, E-8018, C3
4.	SA-516 Grade 70	SA-516 Grade 70		SFA 5.1, E-7018 (b)
~~5.~~	~~SA-508 Class 1~~	~~SA-508 Class 1~~		~~SFA 5.1, E-7018 (b)~~
6.	SA-182 F1	SA-516 Grade 70		SFA 5.1, E-7018
7.	SA-105 Grade II	SA-351 CF8M		SFA 5.11, E-NiCrFe-3 ~~SFA 5.14, ERNiCr-3~~
8.	SA-182 F1	SA-351 CF8M		SFA 5.11, ENiCrFe-3
9.	SA-105 Grade II	SA-182 F316		~~SFA 5.14, ERNiCr-3~~
10.	SB-166	SA-182 F316		Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
11.	SB-167	SA-182 F304		Root SFA 5.14, ERNiCr-3 Remaining SFA 5.11, ENiCrFe-3
12.	SA-516 Grade 70	SA-351 CF8M	~~a.~~ ~~b.~~	~~SFA 5.1, E-7018~~ ~~MIL-E-18193, B-4~~
13.	SA-182 F1	SA-182 F316		SFA 5.11, ENiCrFe-3
14.	SB-166	SA-533 Grade B Class 1		SFA 5.11, ENiCrFe-3
15.	SA-182 Code Case 1334	~~SB-167~~ SB-166		SFA 5.14, ERNiCr-3

a. SFA 5.1, E-7018
b. MIL-E-18193, B4

Table 3
WELD MATERIALS FOR REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS
(Sheet 2 of 2)

	Material Specification	Base Material	Weld Material
16.	SA-516 Grade 70	SA-508 Class 2	SFA 5.5, (a) E-8018, C3
17.	Austenitic stainless steel cladding		SFA 5.9, ER-308 SFA 5.9, ER-309 SFA 5.9, ER-312
18.	Inconel	Inconel	SFA 5.11, ENiCrFe-3 SFA 5.14, ERNiCr-3
19.	SA-182 F-316	SA-508 Class 2	SFA 5.11, ENiCrFe-3
20.	SA-351 CF8M	SA-508 Class 2	SFA 5.11, ENiCrFe-3

When welding SB-166 N06690 or SB-167, SB-06690 base materials, ERNiCrFe-7 and ENiCrFe-7 weld materials may be substituted for ERNiCr-3 and ENiCrFe-3.

(a) Special weld wire with low residual elements of copper and phosphorus is specified for the beltline region.
(b) Filler metal used for Field Welds P1OW1 and P1OW2 have been rated with a strength level of 65 ksi per CE Analytical Report CENC-1460.

Attachment 3

To

W3F1-2004-0115

Additional Information Regarding Turbine Overspeed Evaluation

Additional Information Regarding Turbine Overspeed Evaluation

The turbine overspeed evaluation referred to in the November 8, 2004, submittal is Entergy Operations, Inc. (Entergy) calculation ECM03-009, "Turbine Overspeed Evaluation for 3716 MWt Power Uprate."

This Entergy calculation is supported by Siemens/Westinghouse (SWPC) Engineering Study GO NOE21614, Rev 2, "Steam Turbine Uprating Study Report, Entergy Operations Waterford Unit 3 Plant Uprating Retained Components Evaluation," June 16, 2003.

The SWPC study considered the internal volumes of the steam turbine when determining the turbine overspeed, and provided a methodology to determine the additive overspeed due to any additional steam volumes. Using this methodology, the Entergy calculation considered the additional steam volume in extraction steam piping and the water in the respective heaters, considering the failure of a single extraction steam Reverse Current Valve. The Entergy calculation (supported by the SWPC study), provides the basis for the conclusion that "The new turbine-generator configuration has been evaluated and has concluded that under these conditions the maximum rotor speed achieved will be <120% of rated speed. This analysis also assumes a failure of a reverse current valve." The worst case reverse current valve failure results in an overspeed of 119.9551%.