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December 7, 2010

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287,
Renewed Operating Licenses DPR-38, DPR-47, and DPR-55
High Energy Line Break License Amendment Request - Response to Request for Additional Information

References:

1. Letter to the U. S. Nuclear Regulatory Commission from David Baxter, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for HELB events outside of the Containment Buildings; License Amendment Request No. 2008-005," dated June 26, 2008.
2. Letter to the U. S. Nuclear Regulatory Commission from Dave Baxter, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for HELB Events outside of the Containment Building - Unit 2; License Amendment Request No. 2008-006," dated December 22, 2008.
3. Letter to the U. S. Nuclear Regulatory Commission from Dave Baxter, Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, "Proposed License Amendment Request to Revise the Oconee Nuclear Station Current Licensing Basis for High Energy Line Break Events Outside of the Containment Building," License Amendment Request No. 2008-007, dated June 29, 2009.
4. Letter from John Stang, Senior Project Manager, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission to Dave Baxter (Duke), "Request for Additional Information (RAI) Regarding the Licensee Amendment Requests (LARs) for High Energy Line Break Mitigation," dated October 8, 2010.
5. Letter from Dave Baxter, Site Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, to the U. S. Nuclear Regulatory Commission, "Tornado Mitigation License Amendment Request - Response to Request for Additional Information," dated August 31, 2010.
6. Letter from Dave Baxter, Site Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, to the U. S. Nuclear Regulatory Commission, "Tornado Mitigation License Amendment Request - Response to Request for Additional Information," dated June 24, 2010.

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7. Letter from Dave Baxter, Site Vice President, Oconee Nuclear Station, Duke Energy Carolinas, LLC, to the U. S. Nuclear Regulatory Commission, "License Amendment Request to Revise Portions of the Updated Final Safety Analysis Report Related to the Tornado Licensing Basis," dated June 26, 2008.

By letters dated June 26, 2008, December 22, 2008, and June 29, 2009, Duke Energy Carolinas, LLC (Duke Energy) submitted three (3) license amendments that comprise the final License Amendment Request (LAR) for High Energy Line Break (HELB) events outside of containment (Refs: 1, 2, and 3). This LAR revises the current licensing basis regarding HELB mitigation for the Oconee Nuclear Station (ONS).

Duke Energy received a Request for Additional Information (RAI) related to this LAR on July 24, 2009, and provided a response on October 23, 2009. Duke Energy received an additional RAI on October 8, 2010 (Ref. 4). This submittal responds to this RAI.

The engineering design for questions associated with RAI questions 25, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51 and 52 is subject to revision to support field conditions and as-built configuration. The information contained in the Enclosure and provided on the SharePoint represents the latest information available as of the date of this letter. Also, upon acceptance of the RAI responses, the Oconee HELB Report (ONDS-351) will be revised to include the information in the RAI responses. This action will be tracked within Oconee's corrective action program.

If you have any questions in regard to this letter, please contact Stephen C. Newman, Regulatory Compliance Lead Engineer, Oconee Nuclear Station, at (864) 873-4388.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 7, 2010.

Sincerely,

TPGillespie
T. Preston Gillespie, Jr.
Vice President
Oconee Nuclear Station

Enclosure
Attachment

xc w/enclosure/attachment:

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Enclosure

Response to Request for Additional Information

RAI 1 [H]

The description provided in Section 7.3 of the LARs, letdown line break, does not provide the NRC staff with sufficient information necessary to perform an independent dose consequence calculation. To ensure a complete and accurate safety assessment of the proposed LAR, the NRC staff needs to assess the safety significance of all of the changes to the current licensing basis (CLB) parameters used in the letdown line break dose consequence analysis.

Provide additional information describing all of the basic parameters used in the letdown line break off-site and control room dose consequence analyses. For each parameter, please indicate the CLB value, the revised value where applicable, as well as the basis for any changes to the CLB values. Please provide additional information describing all of the basic parameters used for the letdown line break off-site and control room dose consequence analyses. For each parameter, please indicate the CLB value, the revised value where applicable, as well as the basis for any changes to the CLB values.

Duke Energy Response

Letdown line breaks are not classified as a design basis event in the current licensing basis (CLB). The analysis for postulated line breaks in the letdown line is not contained in the current Chapter 15 of the Updated Final Safety Analysis Report (UFSAR). Breaks were postulated in the letdown line based on the high energy line break (HELB) rules established in a letter from the Atomic Energy Commission (AEC) dated 12/12/1972. A terminal end break was postulated at the containment penetration (between the reactor building wall and the outboard containment isolation valve). Other breaks were also postulated in the high energy portion of the piping downstream of the outboard containment valve and upstream of the pressure reducing devices. An operational analysis was performed for these postulated breaks and documented in MDS Report No. OS-73.2, "Analysis of Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station - Units 1, 2, and 3," dated 4/25/1973.

The analysis contained in the MDS report established the licensing basis for the postulated HELBs in the letdown line. One analysis was provided to address all of the postulated breaks in the letdown line. The analysis assumed a complete severance of the 2-1/2 inch letdown line. No operator action was assumed. Reactor coolant was assumed to flow out the break until the isolation valves were automatically closed. It should be understood that a failure of the containment inboard isolation valve was not postulated for this event since one of the postulated break locations was at the penetration upstream of the containment outboard isolation valve. The normal makeup system was assumed to function to delay the time in which the valves would receive an automatic signal to close. Engineered Safeguards (ES) was assumed to actuate by low reactor coolant pressure at 1500 psig. Analysis showed that the letdown pathway would be isolated at approximately 160 seconds after the break. Off-site releases were considered to be within acceptable limits for this accident. The parameters used to determine that off-site doses were within acceptable limits were not provided for the CLB.

Section 7.3 reflects the analysis that was performed to address changes to the HELB selection and mitigation of postulated pipe breaks in the letdown line. Line breaks in the high energy portion of the letdown line downstream of the containment outboard isolation valve were eliminated based on the piping stress analysis. The previously postulated line break upstream of the containment outboard isolation valve (at the reactor building penetration) remains as the only postulated HELB in the letdown line outside containment. A single active failure of a containment inboard isolation valve (xHP-3 or xHP-4) to close following an ES signal is being considered in the new analysis. Off-site doses as well as control room doses were calculated to determine the required time for operator response to mitigate a single active failure to the containment inboard isolation valve and remain within acceptable limits. The acceptable limits that have been imposed are based on Standard Review Plan (SRP) 15.6.2 and Regulatory Guide 1.183.

The basic parameters used in the off-site and control room dose consequence analyses contained in Section 7.3 are:

- The mass and energy release from the letdown break as used in the dose analysis is described in response to question 3.
- The initial fission product concentrations in the reactor coolant are assumed to be at the maximum equilibrium values as permitted by plant technical specifications.
- An additional iodine spike is assumed to occur. The spike is modeled by increasing the equilibrium fission product activity release rate from the fuel by a factor of 500.
- The fraction of iodine assumed to become airborne and released to the environment is equal to the fraction of the reactor coolant flashing into steam in the depressurization process, and the mass of coolant that is already in the vapor phase. The total flash fraction used for the duration of the accident is calculated to be 31.4%, based on a constant enthalpy process.
- The iodine released from the letdown line break is assumed to be 97% elemental and 3% organic.
- All of the noble gas radionuclides released from the letdown line break are assumed to be released to the environment.
- The atmospheric dispersion factor (X/Q) for the exclusion area boundary (EAB) for the 0-2 hour period is assumed to be 2.2E-4 sec/m^3. X/Q values for the outer boundary of the low population zone (LPZ) off-site dose calculations are:

Time Period	LPZ X/Q (sec/m^3)
0 - 8 hours	2.35E-5
8 - 24 hours	4.70E-6
1 - 4 days	1.50E-6
4 - 30 days	3.30E-7

- RG-1.183 breathing rate values are used in determining off-site dose predictions.
- It is assumed that the Control Room operators will start the CRVS Booster Fans within 30 minutes of ES actuation, which occurs about 6 minutes post-accident.
- The control room booster fan intake flow rate is assumed to be 1215 cfm.
- The Unit 1 and 2 Control room was modeled in the analysis. The analysis used a free volume of 86,447 cubic feet for this control room.
- The unfiltered in-leakage into Units 1 and 2 Control room (which is more limiting than Unit 3) is assumed to be 1202 cfm until the outside air booster fan is started by the operator. After the booster fan is started, the in-leakage is assumed to be 0 cfm, which is conservative for this accident scenario. Due to the short break flow duration and large amount of unfiltered in-leakage that has already contaminated the CR during the unpresssurized time period, once the CR is pressurized, dose actually decreases with increased unfiltered in-leakage (and corresponding increased exhaust flow rate).
- The filter efficiencies for the control room intake iodine filters are assumed to be 99% for particulate, 95% for organic, and 99% for elemental iodine.
- There are two air intakes for the control room outside air booster fans. One intake is assumed to supply 55% of the air flow while the other is assumed to supply 45% of the air flow. Only the air intake with the higher flow is assumed to be located within the wind direction window as defined in RG-1.194 from the postulated release point.

- The X/Q values (adjusted for the 55 / 45 airflow split) used for the Control Room are:

Time Period	X/Q (sec/m^3)
0 - 2 hours	9.85E-4
2 - 8 hours	6.88E-4
8 - 24 hours	3.00E-4
24 - 96 hours	2.29E-4
96 - 720 hours	1.84E-4

  Note: The 0-2 hour X/Q is used during the period of maximum activity release as determined by LOCADOSE. The 2-8 hour X/Q is used during the rest of the first 8 hours.

- RG-1.183 is utilized in determining occupancy factors and breathing rate for the control room operators.
- Dose Conversion Factors are based on the Federal Guidance Reports 11 and 12.

RAI 2 [H]

The analysis provided in Section 7.3 of the HELB report in the Unit 3 LAR, for the letdown line break, assumes a double-ended guillotine break of the 2.5-inch letdown line. Please provide additional information describing the basis for the selection of this line to ensure that the most severe radioactive releases have been considered.

Duke Energy Response

Break selection is based on the high energy line break rules. The letdown line break being analyzed is a terminal end line break outside containment between the inboard and outboard containment isolation valves. Pipe stresses were evaluated in the remaining high energy portion of the letdown line. No other locations on the high energy portion of the letdown line were found to meet the break threshold.

RAI 3 [H]

Provide additional information describing the calculated mass flow rate out of the break as well as the total quantity released as used in the dose analysis.

Duke Energy Response

A thermal-hydraulic analysis was performed using the RETRAN-3D computer code to determine the plant response as well as the break flow rate. Piping losses in the letdown piping inside the Reactor Building were considered in the analysis for the break flow rate. The letdown piping resistances inside containment were evaluated for all three units. Unit 1 had the lowest piping resistance of all three units and hence the highest break flow. Using the lowest piping resistance, the initial break flow rate was calculated to be approximately 102 lbm/sec. The break flow rate decreased to a minimum value of approximately 69 lbm/sec when the RCS pressure decreased to the point where ES actuated at approximately 6 minutes. Break flow rate then began to increase until RCS pressure stabilized. The break flow rate then remained relatively constant at approximately 90 lbm/sec until isolated by the operators.

The RETRAN analysis did not identify when operator actions would be taken to isolate the break. Therefore the break flow analysis was extended out in time to approximately 4000 seconds. The dose analysis assumed operator action would be taken to isolate the break within 20 minutes following ES actuation. The break flow rate used in the dose analysis was assumed to continue for approximately 26 minutes. The total quantity released during the 26 minutes of break flow was calculated to be approximately 140,000 lbm. No leakage was assumed through the closed isolation valve in the analysis. It was also assumed that no cooling of the reactor coolant through the letdown cooler and no ambient heat loss occurred through the piping from the RCS to the break location inside the East Penetration Room to maximize the flashing fraction. Time intervals were established to simplify the inputs into the dose analysis. The maximum break flow rate and the minimum RCS mass over the established time intervals used in the dose analysis is provided in the table below.

Letdown Line Break Flow Rate(lbm/min)	RCS Mass (lbm)	Time Interval(hours)
6170	500904	0.000 to 0.006
6107	499248	0.006 to 0.011
6058	497609	0.011 to 0.017
6010	495985	0.017 to 0.022
5964	494376	0.022 to 0.028
5918	492782	0.028 to 0.033
5876	491202	0.033 to 0.039
5832	489637	0.039 to 0.044
5788	488087	0.044 to 0.050
5766	486558	0.050 to 0.056
5693	485199	0.056 to 0.061
5066	483901	0.061 to 0.067
4955	482635	0.067 to 0.072
4890	481390	0.072 to 0.078
4844	480163	0.078 to 0.083
4758	478956	0.083 to 0.089
4705	478024	0.089 to 0.094
4428	478024	0.094 to 0.100
4288	479930	0.100 to 0.106
4377	481765	0.106 to 0.111
4467	483532	0.111 to 0.117
4557	485229	0.117 to 0.122
4640	486854	0.122 to 0.128
4724	488405	0.128 to 0.133
4810	489885	0.133 to 0.139
5001	491294	0.139 to 0.153
5180	494510	0.153 to 0.167

5332	497309	0.167 to 0.181
5457	499709	0.181 to 0.194
5540	501743	0.194 to 0.208
5526	503485	0.208 to 0.222
5545	505312	0.222 to 0.250
5552	508856	0.250 to 0.278
5550	512356	0.278 to 0.434

RAI 4 [H]

Provide additional information describing the initial fission product concentrations in the reactor coolant system (RCS) and the basis for their selection as the maximum equilibrium values permitted by the technical specifications (TSs).

Duke Energy Response

Departure from nucleate boiling (DNB) fuel failures are not postulated for the letdown line break. SRP 15.6.2 states that the initial fission product concentrations in the primary coolant are assumed to be at maximum equilibrium values permitted by technical specifications. Oconee's technical specification limits on RCS activity are, Dose Equivalent Iodine-131(DEI-131) is less than or equal to 1.0 μCi/gm, and gross specific activity is less than or equal to 100/Ē μCi/gm. Ē is the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the activity of the radionuclides in the reactor coolant. The RCS initial activity used in the dose analysis is provided below:

Isotope	Initial RCS Inventory (Ci)	Isotope	Initial RCS Inventory (Ci)	Isotope	Initial RCS Inventory (Ci)
I-131	6.8E+01	Kr-83M	1.1E+02	Xe-131M	1.0E+03
I-132	1.0E+01	Kr-85M	4.9E+02	Xe-133M	1.3E+03
I-133	1.8E+01	Kr-85	4.1E+03	Xe-133	9.0E+04
I-134	9.5E-01	Kr-87	2.7E+02	Xe-135M	1.0E+02
I-135	5.3E+00	Kr-88	8.4E+02	Xe-135	2.6E+03
				Xe-138	1.6E+02

Note: The initial iodine isotopic inventory was reduced to reflect the 31.4% total flashing fraction.

RAI 5 [H]

Provide the results as well as all the necessary inputs required to determine the RCS concurrent iodine spike isotopic appearance rates and total production for the duration of the assumed spike.

Duke Energy Response

In order to maximize the RCS concurrent iodine spike isotopic appearance rates, assumptions were made to maximize iodine removal rates from the RCS. Iodine removal mechanisms considered include:

- RCS leakage (TS limits of identified and unidentified leakage measured at procedural reference conditions)

- Maximum letdown flow rate
- Letdown purification demineralizer in operation throughout the cycle
- Retention in the pressurizer region

The calculated equilibrium removal rates are:

- I-131 equilibrium removal rate is 3.3E+01 Ci per hour
- I-132 equilibrium removal rate is 4.7E+01 Ci per hour
- I-133 equilibrium removal rate is 6.8E+01 Ci per hour
- I-134 equilibrium removal rate is 7.9E+01 Ci per hour
- I-135 equilibrium removal rate is 6.2E+01 Ci per hour

A spike multiplier of 500 is then applied to the equilibrium rate. The concurrent iodine spike appearance rates used in the dose analysis are provided below. Note that these rates have been reduced to reflect the 31.4% total flashing fraction.

- I-131 appearance rate is 5.1E+03 Ci per hour
- I-132 appearance rate is 7.4E+03 Ci per hour
- I-133 appearance rate is 1.1E+04 Ci per hour
- I-134 appearance rate is 1.2E+04 Ci per hour
- I-135 appearance rate is 9.7E+03 Ci per hour

These iodine spike appearance rates are constant, and continue for the duration of the accident (releases from the break are isolated at approximately 26 minutes).

RAI 6 [H]

BACKGROUND:

The HELB report states for the Unit 1 extraction steam system the failure of column G-17 should not result in structural damage that would block the pre-defined repair pathway. Therefore, damage repairs to restore low-pressure service water (LPSW) remain available.

ISSUE:

The phrase in the HELB report "should not result in failure" leaves the possibility that the column would result in failure. Other portions of the HELB report identify a column that could fail and then address the impact of the column failing. The failure of column G-17 may impact the predefined repair cable routing pathway utilized in the damage repair procedure for providing direct current (DC) power to the emergency 4160 volt 0/) switchgear. However, an alternate pathway for the cable routing remains available to effect repairs.

REQUEST:

Provide a description of the consequences of the failure of Column G-17 and the impact on the plant's ability to restore the LPSW system.

Provide the damage repair procedure that addresses the potential consequences of the failure of Column G-17.

Duke Energy Response

The consequences of the failure of Turbine Building Column G-17 due to postulated HELB 1ES-020-R-5 are documented in Calculations OSC-7516.09 and OSC-7516.10 (HELB Report ONDS-351, Revision 2 – References 10.2.12 & 10.2.13) and summarized on Page 4 of Table 4.2-3 of the

HELB Report. There are two (2) types of adverse interactions that result from the failure of Column G-17. These include:

- Loss of Shutdown Equipment (Collateral Damage) due to the interaction of the column and/or generated structural & equipment debris
- The indirect loss of Shutdown Equipment caused by the resulting Turbine Building flood

The Shutdown Equipment adversely affected by the direct interaction with the failed Column G-17 and/or generated structural or equipment debris is:

- Loss of the Unit 1 Main Condenser "1A" & subsequent loss of inventory from the hotwell. This results in a loss of the Unit 1 EFW suction source.
- Rupture of the 78 inch Condenser Circulating Water (CCW) pipe line, containing CCW Valve 1CCW-14, at the connection to the Unit 1 Main Condenser
- Rupture of the 6 inch Emergency Feedwater (EFW) pipe line between valve 1FDW-313 and the Auxiliary Building and downstream of valve 1FDW-373. This rupture prevents the feeding of the "1A" Steam Generator from any Unit 1 EFW source and from any cross connection sources.
- Loss of cable trays B-100b, B-109, B-109a, and B-144
- Loss of cable tray A-102a and subsequent loss of the "B" LPSW Pump
- Loss of electrays from cable trays A-102a, A-111, and A-113b

The Shutdown Equipment indirectly adversely affected from the Turbine Building flooding includes:

- Emergency Feedwater Pumps (all units)
- Low Pressure Service Water (LPSW) Pumps (all units)
- Engineered Safeguards (all units)

The identified Shutdown Sequence for postulated HELB 1ES-020-R-5 is provided on Pages 4-28 4-29, 4-45, & 4-52 of the HELB Report, and the description of the collateral damage is provided on Page 8-10 of the HELB Report. Since the resulting Turbine Building flood causes the loss of the LPSW pumps on each unit, replacement of the pump motors is required in order to re-establish the functionality of the LPSW System. There are pre-defined pathways for the replacement of these pump motors. If the primary pathway is blocked, due to the structural debris from the postulated failure of Column G-17, repairs to the LPSW pumps would be accomplished by using an alternate pathway away from the debris area or delayed until debris is cleared.

Page 8-10 of the HELB Report (ONDS-351) will be revised to be consistent with the response to this RAI.

The damage repair procedures utilized to restore the LPSW and CCW Systems are the following:

- Emergency Plan Procedure, RP/0/B/1000/022 – Procedure for Major Site Damage Assessment and Repair, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/024 – Plant Assessment and Alignment Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/025 – Cooldown Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, IP/0/A/0050/002 – Site Damage Control Procedure, ONS Units 1, 2, & 3
- Emergency Procedure, MP/0/A/3009/012 - Emergency Plan for Replacement of HPI, LPI, and LPSW Motors Following Damage in Turbine Building or Auxiliary Building, ONS Units 1, 2, & 3.

These procedures (References 10.3.21 through 10.3.25, respectively, in the ONS UFSAR HELB Report (ONS-351)), are referenced within the text of the HELB Report as used in the Shutdown Sequence for the postulated HELBs. These procedures have been revised to incorporate the restoration of the LPSW and the CCW Systems following postulated HELBs in the Turbine Building, which result in potential obstruction of the primary pathway to the equipment. These procedures will be uploaded to the SharePoint site when completed.

RAI 7 [H]

BACKGROUND:

The HELB report stated that the Unit 1 main feedwater system states the following:

- *Sub-break 9 interacts with Column G-23 and may result in its failure.*
- *The failure of Column G-23 is not expected to block the pre-defined repair pathway to replace the LPSW pump motors.*

ISSUE:

The phrase "is not expected to block the pre-defined repair pathway," leaves the possibility that the failure of column G-23 would block the pre-defined repair pathway. Other portions of the HELB report identify a column that could fail and then addresses the impact of the column failing. For example, for the Unit 2 main feedwater system the failure of the column may impact the pre-defined repair cable routing pathway utilized in the damage repair procedure for providing DC power to the emergency 4160V switchgear. However, an alternate pathway for the cable routing remains available to effect repairs.

REQUEST:

Provide a description of the consequences of the failure of column G-23 and the impact on the ability to restore the LPSW system.

Provide the damage repair procedure that addresses the potential consequences of the failure of column G-23.

Duke Energy Response

The consequences of the failure of Turbine Building Column G-23 due to postulated HELB 1FDW-031-R-9 are documented in Calculations OSC-7516.09 and OSC-7516.10 (HELB Report ONDS-351, Revision 2 – References 10.2.12 & 10.2.13) and summarized on Page 4 of Table 4.2-4 of the HELB Report. There are two (2) types of adverse interactions that result from the failure of Column G-23. These include:

- Loss of Shutdown Equipment (Collateral Damage) due to the interaction of the column and/or generated structural & equipment debris
- The indirect loss of Shutdown Equipment caused by the resulting Turbine Building flood

The Shutdown Equipment adversely affected by the direct interaction with the failed Column G-23 and/or generated structural or equipment debris is:

- Loss of all Unit 1 EFW System due to damage to the Upper Surge Tank suction source
- Loss of the "B" LPSW Pump due to adverse interactions with Cable Tray A-102a

The Shutdown Equipment indirectly adversely affected from the Turbine Building flooding includes:

- Emergency Feedwater Pumps (all units)
- Low Pressure Service Water (LPSW) Pumps (all units)
- Engineered Safeguards (all units)

The identified Shutdown Sequence for postulated HELB 1FDW-031-R-9 is provided on Pages 4-31, 4-32, 4-46, & 4-53 of the HELB Report. The description of the collateral damage is provided on

Page 8-11 of the HELB Report. Since the resulting Turbine Building flood causes the loss of the LPSW pumps on each unit, replacement of the pump motors is required in order to re-establish the functionality of the LPSW System. There are pre-defined pathways for the replacement of these pump motors. If the primary pathway is blocked, due to the structural debris from the postulated failure of Column G-23, repairs to the LPSW pumps would be accomplished by using an alternate pathway away from the debris area or delayed until debris is cleared.

Page 8-11 of the HELB Report (ONDS-351) will be revised to be consistent with the response to this RAI.

The damage repair procedures utilized to restore the LPSW and CCW Systems are the following:

- Emergency Plan Procedure, RP/0/B/1000/022 – Procedure for Major Site Damage Assessment and Repair, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/024 – Plant Assessment and Alignment Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/025 – Cooldown Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, IP/0/A/0050/002 – Site Damage Control Procedure, ONS Units 1, 2, & 3
- Emergency Procedure, MP/0/A/3009/012 - Emergency Plan for Replacement of HPI, LPI, and LPSW Motors Following Damage in Turbine Building or Auxiliary Building, ONS Units 1, 2, & 3

These procedures are References 10.3.21 – 10.3.25, respectively, in the ONS UFSAR HELB Report (ONS-351) and these procedures are referenced within the text of the HELB Report as used in the Shutdown Sequence for the postulated HELBs. These procedures have been revised to incorporate the restoration of the LPSW and the CCW Systems following postulated HELBs in the Turbine Building that result in potential obstruction of the primary pathway to the equipment. These procedures will be uploaded to the SharePoint site when completed.

RAI 8 [H]

BACKGROUND:

The HELB report states that the Unit 1 main feedwater system states the following:

- *Sub-break 7 interacts with Column Ga-24 and may result in its failure however collateral damage from the failure of Column Ga-24 may result in a rupture to the circulating cooling water (CCW) piping leading to the turbine building flooding.*
- *The failure of Column Ga-24 may result in debris blocking the pre-defined repair pathway to replacement of the A LPSW pump motor. The pathway to the C LPSW pump remains available. Plant damage repair procedures will need to be revised to include the option of replacing and re-powering the C LPSW pump motor.*

ISSUE:

ONS does not have procedures in place that direct the plant staff to perform actions that may be required in the event Column Ga-24 fails as a result of sub-break 7.

REQUEST:

Provide a description of the consequences of the failure of Column Ga-24. Provide the damage repair procedure that addresses the potential consequences of the failure of Column Ga-24.

Duke Energy Response

The consequences of the failure of Turbine Building Column Ga-24 due to postulated HELB 1FDW-031-R-7 are documented in Calculations OSC-7516.09 and OSC-7516.10 (HELB Report ONDS-351, Revision 2 – References 10.2.12 & 10.2.13) and summarized on Page 3 of Table 4.2-4 of the HELB Report. There are two (2) types of adverse interactions that result from the failure of Column G-23. These include:

- Loss of Shutdown Equipment (Collateral Damage) due to the interaction of the column and/or generated structural & equipment debris
- The indirect loss of Shutdown Equipment caused by the resulting Turbine Building flood

The Shutdown Equipment adversely affected by the direct interaction with the failed Column Ga-24 and/or generated structural or equipment debris is:

- Loss of the 1A MDEFW Pump from failure of valve LPSW-516
- Loss of LPSW essential Header "B" & Valve 1LPSW-139
- Loss of the "B" LPSW Pump

The Shutdown Equipment indirectly adversely affected from the Turbine Building flooding includes:

- Emergency Feedwater Pumps (all units)
- Low Pressure Service Water (LPSW) Pumps (all units)
- Engineered Safeguards (all units)

The identified Shutdown Sequence for postulated HELB 1FDW-031-R-7 is provided on pages 4-31, 4-32, 4-46, 4-52, & 4-53 of the HELB Report, and the description of the collateral damage is provided on Page 8-11 of the HELB Report. Since the resulting Turbine Building flood causes the loss of the LPSW pumps on each unit, replacement of the pump motors is required in order to re-establish the functionality of the LPSW System. There are pre-defined pathways for the replacement of these pump motors. If the primary pathway is blocked, due to the structural debris from the postulated failure of Column Ga-24, repairs to the available LPSW pumps (1A &1C) would be accomplished by using an alternate pathway away from the debris area or delayed until debris is cleared.

The damage repair procedures utilized to restore the LPSW and CCW Systems are the following:

- Emergency Plan Procedure, RP/0/B/1000/022 – Procedure for Major Site Damage Assessment and Repair, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/024 – Plant Assessment and Alignment Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/025 – Cooldown Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, IP/0/A/0050/002 – Site Damage Control Procedure, ONS Units 1, 2, & 3
- Emergency Procedure, MP/0/A/3009/012 - Emergency Plan for Replacement of HPI, LPI, and LPSW Motors Following Damage in Turbine Building or Auxiliary Building, ONS Units 1, 2, & 3

These procedures (References 10.3.21 through 10.3.25, respectively, in the ONS UFSAR HELB Report (ONS-351)), are referenced within the text of the HELB Report as used in the Shutdown Sequence for the postulated HELBs. These procedures have been revised to incorporate the restoration of the LPSW and the CCW Systems following postulated HELBs in the Turbine Building that result in potential obstruction of the primary pathway to the equipment. The revisions to these damage repair procedures (HELB Report ONDS-351, Revision 2 through References 10.3.21 to 10.3.25) also include the use of the Units 1 & 2 LPSW "C" Pump in the procedures, as an alternate means of restoring the functionality of the LPSW System. These procedures will be uploaded to the SharePoint site when completed.

Page 8-11 of the HELB Report (ONDS-351) will be revised to remove the ambiguous statement of needing to revise procedures. The revised statement will be consistent with this RAI response.

RAI 9 [H]

BACKGROUND:

The HELB report in the Unit 3 LAR states the following for the Unit 1 main feedwater system:

Sub-break 11 interacts with Column K-23 and may result in its failure. The failure of Column K-23 is not expected to block the pre-defined repair pathways for LPSW pump motor replacement or the pre-defined repair pathway for cable routing to the low-pressure injection (LPI) and LPSW pump motors.

ISSUE:

The phrase "not expected to block the pre-defined repair pathway," leaves the possibility that the column would block the pre-defined repair pathway if the column fails. Other portions of the HELB report identified columns that could fail and then addressed the impact on their failure. The HELB report is not clear on the pre-defined repair pathway following an HELB which results in the failure of Column K-23.

REQUEST:

Provide a description of the consequences of the failure of Column K-23 and the impact on the plant's ability to restore the LPSW system.

Provide the damage repair procedure to address the potential consequences of the failure of Column K-23.

Duke Energy Response

The consequences of the failure of Turbine Building Column K-23 due to postulated HELB 1FDW-031-R-11 are documented in Calculations OSC-7516.09 and OSC-7516.10 (HELB Report ONDS-351, Revision 2 – References 10.2.12 & 10.2.13) and summarized on Page 4 of Table 4.2-4 of the HELB Report. There are two (2) types of adverse interactions that result from the failure of Column K-23. These include:

- Loss of Shutdown Equipment (Collateral Damage) due to the interaction of the column and/or generated structural and equipment damage
- The indirect loss of Shutdown Equipment caused by the resulting Turbine Building flood

The Shutdown Equipment adversely affected by the direct interaction with the failed Column K-23 and/or generated structural or equipment debris is:

- Rupture of a 42 inch CCW pipe line and causing a Turbine Building flood
- Loss of the Unit 1 EFW suction inventory due to the rupture of several Condensate System pipes
- Loss of Panel 1LS1, LOCA Load Shed Relay Panel
- Loss of the 4160 VAC Bus 1TE Switchgear

- Loss of the "A" Chiller Control Panel causing the loss of redundancy of the Unit 1 & 2 Control Room cooling
- Cable Trays 1ENI3418 & 1ENI3422 associated with pressure switches that provide input to the RPS circuitry

The Shutdown Equipment indirectly adversely affected from the Turbine Building flooding includes:

- Emergency Feedwater Pumps (all units)
- Low Pressure Service Water (LPSW) Pumps (all units)

The identified Shutdown Sequence for postulated HELB 1FDW-031-R-11 is provided on Pages 4-31, 4-32, 4-46, & 4-53 of the HELB Report, and the description of the collateral damage is provided on Pages 8-11 & 8-12 of the HELB Report. Since the resulting Turbine Building flood causes the loss of the LPSW pumps on each unit, replacement of the pump motors is required in order to re-establish the functionality of the LPSW System. There are pre-defined pathways for the replacement of these pump motors. If the primary pathway is blocked, due to the structural debris from the postulated failure of Column K-23, repairs to the LPSW pumps would be accomplished by using an alternate pathway away from the debris area or delayed until debris is cleared.

Pages 8-11 and 8-12 of the HELB Report (ONDS-351) will be revised to be consistent with the RAI response.

The damage repair procedures utilized to restore the LPSW and CCW Systems are the following:

- Emergency Plan Procedure, RP/0/B/1000/022 – Procedure for Major Site Damage Assessment and Repair, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/024 – Plant Assessment and Alignment Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, OP/0/A/1102/025 – Cooldown Following Major Site Damage, ONS Units 1, 2, & 3
- Emergency Procedure, IP/0/A/0050/002 – Site Damage Control Procedure, ONS Units 1, 2, & 3
- Emergency Procedure, MP/0/A/3009/012 - Emergency Plan for Replacement of HPI, LPI, and LPSW Motors Following Damage in Turbine Building or Auxiliary Building, ONS Units 1, 2, & 3

These procedures (References 10.3.21 through 10.3.25, respectively, in the ONS UFSAR HELB Report (ONS-351)), are referenced within the text of the HELB Report as used in the Shutdown Sequence for the postulated HELBs. These procedures have been revised to incorporate the restoration of the LPSW and the CCW Systems following postulated HELBs in the Turbine Building that result in potential obstruction of the primary pathway to the equipment. These procedures will be uploaded to the SharePoint site when completed.

RAI 10 [H]

BACKGROUND:

The HELB report in the Unit 3 LAR states the following for the Unit 2 condensate system:

The failure of either column (H-32 or H-33a) may impact the pre-defined repair cable routing pathway utilized in the damage repair procedures. An alternate pathway must be determined.

ISSUE:

The phrase "an alternate pathway must be determined" leaves open the question of what will ONS do if there is a column failure.

REQUEST:

Explain why the damage control procedures do not identify the alternate pathway.

Provide a description of what actions will be taken to prepare for the possibility of the loss of the pre-defined repair pathway.

Duke Energy Response

Prior to the Unit 3 HELB LAR release date, the plant damage assessment and repair procedures had not been reviewed for all HELB scenarios that may impact pre-defined motor/cable replacement pathways. Plant walkdowns have since been performed and alternate pathways or plant modifications have been identified for all relevant HELB scenarios.

Plant damage assessment and repair procedures are being revised to address alternate cable/motor replacement pathways for HELB scenarios that may affect the pre-defined repair pathways (see responses to RAIs 6 – 9 for list of procedures). After revision, these procedures will be uploaded to the Shareware site for NRC review. The related plant modifications are committed items as described in commitments 30H, 38H and 44H.

RAI 11 [H]

BACKGROUND:

The HELB report in the Unit 3 LAR states the following for the Unit 2 extraction steam system:

Running break 2-ES-024-R is on 42-inch 2C steam extraction pipe. The failure of the column may impact the pre-defined repair pathway for motor replacement and power cable routing for the 3A LPSW Pump, should it be needed.

ISSUE:

The phrase "may impact the pre-defined pathway" leaves the possibility that the failure will impact the pre-defined pathway.

REQUEST:

Provide a description of what actions will be taken to prepare for the possibility of the loss of the pre-defined repair pathway.

Duke Energy Response

Prior to the Unit 3 HELB LAR release date, the plant damage assessment and repair procedures had not been reviewed for all HELB scenarios that may impact pre-defined motor/cable replacement pathways. Plant walkdowns have since been performed and alternate pathways or plant modifications have been identified for all relevant HELB scenarios.

Plant damage assessment and repair procedures are being revised to address alternate cable/motor replacement pathways for HELB scenarios that may affect the pre-defined repair pathways (see responses to RAIs 6 – 9 for list of procedures). After revision, these procedures will be uploaded to the Shareware site for NRC review. The related plant modifications are committed items as described in commitments 30H, 38H and 44H.

RAI 12 [H]

BACKGROUND:

The HELB report states the following for the Unit 2 extraction steam system:

"Running break 2-ES-20-R is on a 36-inch..."

"Sub-break 3 interacts with Column H-39 and may result in its failure. The failure of the column may impact the pre-defined repair pathway for motor replacement and power cable routing for the 3A LPSW Pump, if needed."

ISSUE:

For the potential failure of Column H-39, the HELB report is unclear on the pre-defined repair pathway.

REQUEST:

Provide a description of what actions will be taken for the possibility of the loss of the predefined pathway.

Duke Energy Response

Prior to the Unit 3 HELB LAR release date, the plant damage assessment and repair procedures had not been reviewed for all HELB scenarios that may impact pre-defined motor/cable replacement pathways. Plant walkdowns have since been performed and alternate pathways or plant modifications have been identified for all relevant HELB scenarios.

Plant damage assessment and repair procedures are being revised to address alternate cable/motor replacement pathways for HELB scenarios that may affect the pre-defined repair pathways (see responses to RAIs 6 – 9 for list of procedures). After revision, these procedures will be uploaded to the Shareware site for NRC review. The related plant modifications are committed items as described in commitments 30H, 38H and 44H.

RAI 13 [H]

The licensee's response to RAI 2 did not include a commitment to meet all of the criteria in Branch Technical Position (BTP) Mechanical Engineering Branch (MEB) 3-1, Revision 2, as requested in part (a) of RAI 2, despite the continued application of portions of the BTP. Nor did the response provide a detailed comparison of the full criteria contained in BTP MEB 3-1 with the ONS proposed LAR HELB criteria, as requested in Part (b).

a) Please provide a detailed comparison of the full criteria contained in BTP MEB 3-1 with the ONS proposed LAR HELB criteria.

   The licensee's criteria for defining pipe break locations lacks the provisions found in the Giambusso Letter that breaks are defined where thermal stresses alone exceed $0.8S_A$.

b) Provide a commitment to define breaks where thermal stresses alone exceed $0.8S_A$, or justify not doing so.

   The licensee's response to RAI 2 states in the discussion of postulation of critical cracks:

   "A further enhancement is provided for portions of the MS [main stream] and MFDW [main feedwater] Systems located in the [Auxiliary Building] AB. These systems receive periodic volumetric inspections at all accessible girth welds locations and adjacent base metal to provide early warning of potential degradation in these systems that might result in the

formation of a break or crack."

c) Please clarify what is being done to assure the integrity at inaccessible girth weld locations and provide specific data identifying the locations and number of inaccessible locations.

Duke Energy Response

The current licensing basis for the Oconee Station for High Energy Line Breaks is the Giambusso/ Schwencer letters. Oconee is proposing adoption of elements of the Standard Review Plan (SRP) and BTP MEB 3-1, where practical, or where clarification of licensing approaches is available.

(a) The requested information is given in the Attachment to this submittal.

(b) In the absence of primary stress, secondary stress, such as thermal, is a poor predictor of potential pipe failure locations. Primary stress is needed to cause a potential pipe failure. The ASME Code Section NB-3213.8 (1977 edition) defines primary stress as follows: "Any normal or a shear stress developed by an imposed loading which is necessary to satisfy the laws of equilibrium of external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. Primary stresses which considerably exceed the yield stress will result in failure or, at least, in gross distortion."

ASME Code NB-3213.13 defines thermal stress as follows: "Thermal stress is a self balancing stress produced by a non-uniform distribution of temperature or by differing thermal coefficients of expansion. Thermal stress is developed in a solid body whenever a volume of material is prevented from assuming the size and shape that it normally should under a change in temperature."

In section NB-3213.13(b), the Code notes: "Local thermal stress is associated with almost complete suppression of the differential expansion and thus produces no significant distortion. Such stresses shall be considered only from the fatigue standpoint and are therefore classified as local stresses in Table NB-3217-1." Since thermal stress is self balancing, thermal stress which exceeds the yield stress will not result in failure. Repeating cycles of thermal stress exceeding the yield stress may result in cracking due to fatigue, however, the potential for critical crack formation is addressed by the postulation of critical cracks where the actual stress exceeds the crack stress threshold of $.4 \times (S_A + S_H)$.

Giambusso/Schwencer included the requirement to postulate break locations where the actual stress exceeded $.8S_A$. However, BTP MEB 3-1 includes no such requirement. Duke Energy concluded that the omission of the thermal stress threshold in BTP MEB 3-1 is recognition by the regulatory authorities that thermal stress, in the absence of primary stress, cannot cause pipe rupture failures.

(c) Inaccessible girth welds are enclosed by the MFDW guard pipes adjacent to Reactor Building penetrations 25 & 27. The guard pipes form part of the MFDW rupture restraints as described in Section 8 (item 5) of ONDS-351. The inaccessible girth welds are present in Units 1 & 2, but not in Unit 3. For Units 1 & 2, the MFDW A header(s) include an 18 degree elbow located just upstream of RB penetration 27 and the MFDW rupture restraint. While the upstream girth weld of the 18 degree elbow is accessible and volumetrically inspected once during each 10 year ASME Section XI in-service inspection interval, the downstream girth weld is enclosed by the aforementioned guard pipe. Similarly, the Units 1 & 2 MFDW B header(s) include a 32 degree elbow located just upstream of the RB penetration 25 and the MFDW rupture restraint. Again, the upstream girth weld of the elbow is accessible and volumetrically inspected once during each 10 year ASME Section XI in-service inspection interval, the downstream girth weld is enclosed by the aforementioned guard pipe. The Unit 3 headers contain no such elbows, and as such there are no girth welds enclosed by the MFDW rupture restraint guard pipe.

As described in ONDS-351 (Section 8, Item 5), each MFDW guard pipe encloses the postulated MFDW break location(s). Since these downstream elbow girth welds are adjacent to the postulated break location inside the guard pipe, assuming a break at the inaccessible weld(s) would result in no greater consequences than those that would occur for break(s) postulated inside the guard pipe.

RAI 14 [H]

Based on the licensee's response to RAI 3 (items a), b), and c)), the following additional item is requested with regards to RAI 3:

Please provide the following information:

For every system classified as moderate-energy based on the 1 percent of total plant run time, provide the total time spent at high-energy conditions and the total time spent operating. In addition, provide the total time the plant ran during the same time interval.

These times can be taken from the time interval researched in the previous RAI submittal:

- For Unit 1: From 7/8/1999 to 6/1/2008
- For Unit 2: From 12/16/1999 to 12/12/2008
- For Unit 3: From 5/21/2000 to 11/11/2008

Duke Energy Response

In accordance with ONDS-351, Analysis of Postulated HELBs Outside of Containment, Duke does not postulate pipe ruptures or "critical cracks" in high energy lines that operate at high energy conditions less than 1% of the total plant (unit) operating time (Normal Plant Conditions). Normal Plant Conditions have been defined as operation in Modes 1, 2, 3 and 4.

Systems excluded using the 1% criterion were emergency systems. The emergency systems include, Emergency Feedwater (EFW), Reactor Building Spray (RBS), the 'C' High Pressure Injection (HPI) pump discharge, and the Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW). These systems are operated following plant emergencies or for surveillance testing. When these systems are operating, they always operate in the high energy state. Therefore, normal plant startup and shutdown sequences and the associated times spent in the different modes do not determine the time the emergency systems are exposed to high energy conditions. A time interval of 1581 days (from 1/1/2005 to 5/1/2009) was selected to provide a representative historical period for review of the various systems. The time interval was judged to be of sufficient duration to reflect typical high energy operating times.

1% Exclusions – Time Spent in High Energy (1/1/2005 to 5/1/2009)

	Unit 1 (days)	Unit 2 (days)	Unit 3 (days)
'A' Motor Driven EFW Pump Discharge	2.9	1.8	1.1
'B' Motor Driven EFW Pump Discharge	2.6	1.3	1.2
Turbine Driven EFW Pump Discharge	2.5	2.9	3.1
'A' RBS Pump Discharge	0.9	0.8	0.7
'B' RBS Pump Discharge	0.8	0.7	0.7
'C' HPI Pump Discharge	1.9	1.5	0.9

The SSF ASW is an emergency system that supports all three units. The SSF ASW Pump discharge was operated in a high energy condition for approximately 3.2 days during the same time interval of 1581 days (from 1/1/2005 thru 5/1/2009).

The total operating time spent in Modes 1 through 4 for each unit within the time interval from 1/1/2005 to 5/1/2009 is provided below:

- Unit 1 Total Operating Time in Modes 1 through 4 was approximately 1440 days.
- Unit 2 Total Operating Time in Modes 1 through 4 was approximately 1480 days.
- Unit 3 Total Operating Time in Modes 1 through 4 was approximately 1500 days.

RAI 15 [H]

For the licensee's response to RAI 4(b), please include within the body of the text, references to the specific documentation that supports the discussion of field walk downs, piping interactions, break analyses, and mitigation of piping break effects.

Duke Energy Response

The report ONDS-351 includes reference 10.3.17, "HELBs Outside Containment Walkdown Criteria & Requirements." This was the procedure used to conduct plant surveys to determine the potential for high energy line breaks to affect safe shutdown target equipment. The locations of HELBs are documented in ONDS-351 reference(s) 10.2.2 (Calculation OSC 7516.01, ONS Unit 1 High Energy Line Break Stress Evaluation), 10.2.39 (Calculation OSC 7517.01, ONS Unit 2 High Energy Line Break Stress Evaluation), and 10.2.52 (Calculation OSC 7518.01, ONS Unit 3 High Energy Line Break Stress Evaluation). The results of the plant surveys are documented in ONDS-351 references 10.2.6 (Calculation OSC 7516.02, ONS - Unit 1 - Pipe Rupture Evaluation HELB Outside Containment Plant Walkdowns), 10.2.40 (Calculation OSC 7517.02, ONS - Unit 2 - Pipe Rupture Evaluation HELB Outside Containment Plant Walkdowns) , and 10.2.53 (Calculation OSC7518.02, ONS - Unit 3 - Pipe Rupture Evaluation HELB Outside Containment Plant Walkdowns). These calculations document the potential for high energy line breaks to affect safe shutdown target equipment and piping. The safe shutdown equipment and piping is documented in ONDS-351 references 10.2.4 (Calculation 8089.01, High Energy Line Break (HELB) Safe Shutdown Target List (SSTL) - ONS Units 1, 2, & 3)) and 10.2.15 (Calculation 8089.02, High Energy Line Break (HELB) Safe Shutdown Target List (SSTL) Pressure Boundary Piping (ONS Units 1, 2, & 3)).

These calculations do not provide the mitigation strategy for each documented high energy line break and the potentially affected safe shutdown target equipment and piping. As stated within ONDS-351 Section 1.1, the analysis of high energy line break interaction(s) and the pathways to Safe Shutdown / Cold Shutdown is based upon the station configuration following the completion of HELB modifications described in Section 9.0 of the report. The analysis of the mitigation of high energy line breaks is contained in Section 4 (Unit 1), Section 5 (Unit 2) and Section 6 (Unit 3) respectively of ONDS-351.

RAI 16 [H]

In response to RAI 5, the licensee judged that the energy contained in the 1.5-inch and 2-inch high energy (HE) piping as being insufficient to damage adjacent piping systems or structural components. Please identify the technical evaluation or reference that supports this basis?

In addition,

a) How many feet of HE piping in excess of a 1 nominal pipe size (nps) lower limit are there in the plants? List lengths for each size in excess of 1 nps.

b) Are there any restrictions between the HE piping in excess of 1 nps and the ultimate gas source?

c) The Electro Hydraulic Control systems are not mentioned in the detailed response. Please discuss why this piping has been eliminated from the HE candidates?

Duke Energy Response

This RAI is related to the nitrogen and Electro-Hydraulic Control (EHC) systems.

The nitrogen system consists of ten horizontal supply tanks located outside the Turbine Building. These tanks are normally pressurized to approximately 2000 psig. These tanks supply the nitrogen headers located inside the Turbine Building and the Auxiliary Building. Supply from the tanks is provided by ¾ inch nominal pipe size (nps) piping to a pressure reducing valve that regulates the downstream piping pressure to approximately 625 psig. The pressure reducing valve is also located outside the Turbine Building. The downstream piping increases in size to 1.5 inch nps piping before it enters the Turbine Building. There is approximately 220 feet of the 1.5 inch nps high pressure nitrogen piping inside the Turbine Building. The high pressure nitrogen supply piping then reduces in size to 1 inch nps. The 1 inch nitrogen supply piping is routed to the Auxiliary Building from the Turbine Building. There are two locations inside the Auxiliary Building on the 2nd Floor Hallway where the high pressure nitrogen piping increase in size from 1 inch nps to 2 inches nps. Any ruptures in the 1.5 inch or 2 inch nitrogen piping would be limited by the ¾ inch pressure reducing valve located outside the Turbine Building.

The EHC system is located in the Turbine Building basement for each unit. The EHC system is an oil system that is normally pressurized to approximately 1600 psig. There are two EHC pumps per unit that can provide the source for high pressure. Only one pump per unit is normally operating. Each pump has a design flow rate of 53 gpm at a discharge pressure of 1600 psig. The largest size of high pressure piping in the EHC system is 1.5 inch nps. Any ruptures in the 1.5 inch EHC piping would be limited by the capacity of the EHC pumps.

Breaks were not postulated in the EHC system due to the limited potential for breaks in the system to cause pipe whips and jet impingement loads that could severely damage systems, structures, and components necessary to safely achieve a safe shutdown state. Since the system operates at less than the saturation point of the fluid, the discharge jet is characterized by a nearly constant jet approximately equal to the break diameter. In addition, given the substantial viscosity of the EHC fluid, as compared to water, the friction losses in the system are greater than that for comparable steam, saturated water, and sub-cooled water systems. This increase in frictional losses will result in a smaller steady state thrust coefficient, limiting the magnitude of the discharge jet. The combination of these factors, along with the relative small size of the EHC system piping, provides reasonable assurance that a break in the EHC system will not adversely affect systems, structures, and components in the vicinity.

RAI 17 [H]

In response to RAI 6(a), the licensee states that "Giambusso/Schwencer does address the postulation of terminal end breaks at isolation valves for Class 1 piping."

a) Provide and fully reference specific statements from criteria that discuss isolation valves and, for each instance/location where such end breaks are postulated, demonstrate compliance with the criteria.

The licensee states that "Giambusso/Schwencer required the postulation of terminal end breaks at rigidly fixed valves that may act to restrain thermal movement." The licensee also states, "...all isolation valves that serve in this manner are in line valves that are not independently supported or supported in a way that would prohibit piping motion and thermal movement."

b) Please provide documentation and a technical basis that confirms this condition at all isolation valves.

The licensee states that relative to postulating break locations, "The NRC has previously approved this interpretation at the Oconee Nuclear Station for the Passive Low Pressure Injection Cross Connection Modifications."

c) Document similarities which demonstrate applicability to the current ONS LAR on excluding boundary valves from being terminal ends.

Duke Energy Response

Response to 17(a):

Section 2.2.2 of ONDS-351 addresses the postulation of terminal end breaks at isolation valves. It states:

"The postulation of terminal end breaks at the first normally closed valve(s) separating portions of a system maintained pressurized during normal operations and portions of a system not maintained pressurized depends on whether the system has a seismic analysis that is continuous across the valve. For systems or portions of systems that are not seismically analyzed, breaks are postulated to occur at all piping girth welds in the system including those that attach to normally closed valves. For systems or portions of systems that are seismically analyzed, and the analysis is continuous across the normally closed valve, such that stress can be accurately determined, break and crack locations are determined based on comparison to the break and crack stress thresholds."

Giambusso/Schwencer specifies in 2(a) the criteria for the postulation of break locations in each piping run. 2(a)(1) specifies that breaks be postulated at terminal ends. Footnote (3), referenced in 2(a), notes, "A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement." Duke Energy interprets this to mean that break locations should be postulated at rigidly fixed valves, since they can act as the terminus to a piping run, and thus act as a terminal end. In addition, Duke Energy interprets this to mean that break locations should be postulated at normally isolated valves, if such valves are rigidly restrained. As provided in the response to RAI 6(a) there are no isolation valves that serve as the boundary between high energy piping and moderate energy piping that are rigidly supported independent of the piping system. As such no terminal end breaks were postulated at isolation / boundary valves in continuous seismically analyzed systems.

Response to 17(b):

ONDS-351 provides the summary of the analysis of pipe breaks postulated in high energy systems outside containment at Oconee Nuclear Station. The postulation of break locations is found in ONDS-351 reference(s) 10.2.2 (Calculation OSC 7516.01, ONS Unit 1 High Energy Line Break Stress Evaluation), 10.2.39 (Calculation OSC 7517.01, ONS Unit 2 High Energy Line Break Stress Evaluation), and 10.2.52 (Calculation OSC 7518.01, ONS Unit 3 High Energy Line Break Stress

Evaluation). These calculations document break locations at normally closed isolation valves if such valves are included in piping systems or portion of piping systems that were not seismically analyzed. These calculations also document break and crack locations for seismically analyzed lines adjacent to normally closed valves, if the actual stress at those locations exceed the break and crack thresholds. In order to determine the effect of a postulated break, plant surveys were completed. The results of the plant surveys are documented in ONDS-351 references 10.2.6 (Calculation OSC 7516.02, ONS - Unit 1 - Pipe Rupture Evaluation HELB Outside Containment Plant Walkdowns), 10.2.40 (Calculation OSC 7517.02, ONS - Unit 2 - Pipe Rupture Evaluation HELB Outside Containment Plant Walkdowns) , and 10.2.53 (Calculation OSC7518.02, ONS - Unit 3 - Pipe Rupture Evaluation HELB Outside Containment Plant Walkdowns).

Response to 17(c):

Section 2.2.2 of ONDS-351 addresses the postulation of breaks and critical cracks at isolation valves, as follows:

"Breaks & Critical Cracks at closed valves are postulated as follows. The postulation of terminal end breaks at the first normally closed valve(s) separating portions of a system maintained pressurized during normal operations and portions of a system not maintained pressurized depends on whether the system has a seismic analysis that is continuous across the valve. For systems or portions of systems that are not seismically analyzed, breaks are postulated to occur at all piping girth welds in the system including those that attach to normally closed valves. For systems or portions of systems that are seismically analyzed, and the analysis is continuous across the normally closed valve, such that stresses can be accurately determined, break and crack locations are determined based on comparison to the break and crack stress thresholds."

By letter dated September 29, 2003, the NRC staff issued Amendments 335, 335, and 336 for Oconee, Units 1, 2, and 3, to support the installation of a passive Low Pressure Injection (LPI) System cross connects inside containment. For all three units, the amendments approved the use of Standard Review Plan (SRP) 3.6.2 Branch Technical Position MEB 3-1.

The March 20, 2003, LAR (approved by Amendments 335, 335, and 336), describes a boundary valve (xLP-47 and 48) that exists in each LPI train that separate the high energy portion of the system from the moderate energy portion of the system. The submittal continued that normally, in cases where a valve constitutes the boundary between moderate and high energy lines, a terminal end break is postulated per BTP MEB 3-1 B.1.c (1) (a) footnote 3. In the circumstances described, the stress analysis of the LPI system is continuous across the boundary, such that the stress levels can be accurately portrayed for each applicable load case, and as such, no terminal end breaks were postulated at the boundary valve(s). It was further described that this treatment meets the intent of footnote 3, since the piping both upstream and downstream of the boundary valves was included in the same stress analysis model. The Staff acknowledged this request, by noting in Section 3.2 (page 9) of the referenced SER, "...that the piping model that includes the valves satisfies the intent of the footnote in that the valves are modeled in the piping run and they are not independently supported in such a way as to represent a terminal end," The staff provided acceptance of this position on page 10 of the SER.

The application of MEB 3-1 described above is the same as requested by the High Energy Line Break LAR as documented in ONDS-351. For comparison purposes, the request to not treat high energy / moderate energy boundary valves in analyzed piping that includes the seismic loading case is given below:

- The boundary valves described in the HELB submittal are in-line valves not independently supported from the piping system.

- The stress analysis of the piping run(s) that includes the boundary valves described in the HELB submittal is continuous on both the downstream and upstream sides of the valves, such that actual stress for each load case is accurately known.

Finally, as stated in ONDS-351, this request only applies to seismically analyzed piping systems or subsystems. For piping not seismically analyzed breaks are postulated at all piping girth welds, including those between system piping and boundary isolation valves.

RAI 18 [H]

The note in BTP MEB 3-1 discussed in RAI 7 and its response states in part:

"A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior." This is a vintage artifact of the limited analysis capabilities available at the time of the BTP. Model sizes were limited to the point where branch piping was included in a model of run piping, where it had no influence, and was not accurately represented in the model's response.

a) What modeling criteria does the licensee have in place to ensure an accurate response of branch piping that does not influence the response of the run piping, e.g., responds in a significantly different frequency range than the run piping?

A note in RAI 7 states:

> *The NRC staff has, in the past, asked the licensee to clarify that it will satisfy the complete criteria contained in Footnote 3 of BTP MEB 3-1. It does not appear that this has taken place in the proposed LAR. In addition, the NRC staff has previously requested the licensee to compare its proposed HELB criteria with the full criteria contained in BTP MEB 3-1 in order for the NRC staff to perform a thorough safety review of the Duke HELB proposal. The proposed LAR only addresses the criteria from BTP MEB 3-1, which provides relaxations to the Oconee licensing basis HELB criteria.*

b) These two requests are noted as still having not been addressed, and, therefore are still pending.

Duke Energy Response

(a) For analysis purposes, branch lines are those having a diameter less than one-fourth of the run pipe diameter, a moment of inertia of less than 1/25th of the run moment of inertia, or a section modulus less than one-tenth that of the run pipe section modulus. Branch lines meeting any of these criterion were decoupled from the run pipe stress analysis model and evaluated as a separate model with the branch point as one of the boundary points.

Thermal analysis of the decoupled branch line included thermal movements of the run pipe applied as anchor movements to the branch line. Similarly, seismic analysis of the decoupled branch line included inertial displacements and/or seismic anchor motion displacements of the run pipe applied as anchor movements to the branch line.

(b) See response to RAI 13(a).

RAI 19 [H]

Seismic Category I piping is piping classified by application of criteria found in Regulatory Guide 1.29, "Seismic Design Classification." Please provide a list of the seismic category I piping, for which the summary information found in item 4 of the Giambusso letter still needs to be provided and address how the requirements from (a) to (e) of item 4 are met.

Duke Energy Response

There are three Seismic Category I piping systems located outside containment, of which a portion is classified as high energy. Those systems are the High Pressure Injection (HPI) System, the Main Steam (MS) System, and the Main Feedwater (MFDW) System. Giambusso/Schwencer item 4 requested "...that a summary be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of the postulated pipe break." In the October 23, 2009 response for RAI 8, we noted that the Giambusso letter, on page 1, included the following:

"Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual basis the applicability of each of the following items for inclusion in their submittals."

In addition it was noted in the October 23, 2009 submittal "that dynamic analyses were performed for the break scenarios that warranted a dynamic analysis." As such, no dynamic analyses were performed for the HPI System for the purposes of designing piping supports and or rupture restraints. The postulated break locations are as shown in ONDS-351, Figures 4.1-7 (Unit 1), 5.1-7 (Unit 2), and 6.1-7 (Unit 3). For the postulated locations, jet impingement forces were determined in accordance with ANSI/ANS 58.2 -1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture". Once the jet impingement forces were determined, plastic hinges were postulated and whip interaction zones established. Following that, surveys were made of the interaction zones to identify any safe shutdown equipment. Identified safe shutdown equipment located within the interaction zones were considered to be damaged and rendered non functional. See response to RAI 15 for further information regarding the evaluation of postulated high energy line breaks.

A similar process was followed for the MS System. Postulated break locations are as shown in ONDS-351, Figures 4.1-8 (Unit 1), 5.1-8 (Unit 2), and 6.1-8 (Unit 3). For the postulated MS break in the East Penetration Room, internal pressurization of the room was determined as described in ONDS-351, Section 8, Item 20.

A similar process was followed for the MFDW System. Postulated break locations are as shown in ONDS-351, Figures 4.1-4 (Unit 1), 5.1-4 (Unit 2), and 6.1-4 (Unit 3). For the postulated MFDW breaks in the East Penetration Room, internal pressurization of the room was determined as described in ONDS-351, Section 8, Item 20.

As stated in ONDS-351 Section 8, item 4, with the exception of the two MFDW rupture restraints, located in the East Penetration Room, evaluations of the effects of whip and jet impingement associated with postulated break locations of Category I piping assumed unrestrained lines. As such, there was no need to determine the dynamic response for breaks in Category I piping, since no supports in the lines were designed to absorb these loads. Rather, the safe shutdown equipment, located in the zone of influence of these breaks were assumed failed and rendered non operational.

The overall mitigation strategy for breaks in Category I systems is the availability of other equipment remote from the postulated break locations that could be used to bring the affected unit to a safe shutdown state. Thus for postulated breaks in Category I systems located in the Auxiliary Building, systems and equipment located in the Turbine Building and or the Standby Shutdown Facility (SSF) would be available to mitigate the break event. Similarly, for postulated breaks in Category I systems located in the Turbine Building, systems and equipment located in the Auxiliary Building and or the SSF would be available to mitigate the break event.

RAI 20 [H]

This question is applicable to the licensee's responses to RAIs 9 and 10.

How does the criteria and methodology used to design containment penetrations under line rupture forces and moments generated by their own rupture compare to the criteria and methodology used with pipe whip restraints in the current HELB evolution?

Duke Energy Response

The design pressure and temperature of the piping systems penetrating containment are used to determine the line rupture forces and moments caused by their own rupture applied to the containment penetration(s). The normal operating pressure and temperature of the high energy systems are used to determine the line rupture forces and moments caused by the postulation of break locations.

RAI 21 [H]

The LAR identifies various sections of the high energy piping that have been excluded due to normal operating temperature and pressure conditions.

Please provide a complete list of excluded high energy piping systems, if such a list cannot be found in the calculation.

Duke Energy Response

High energy lines were excluded from the evaluations if it was shown that the normal operating conditions were below the threshold for high energy. In addition, piping downstream of a normally closed valve in a high energy system was excluded. The normal operating configuration for a high energy system was based on the operating configuration at 100% rated full power. High energy systems that are not normally in operation were excluded. Systems that operate at or below atmospheric pressure were also excluded from the evaluations since these systems were judged to have insufficient energy to create pipe whips or jets. Finally, oil and gas piping were not considered high energy systems since they possess limited energy (see response to RAI 16).

Based on the above information, the following systems or portions of systems were excluded from the evaluation of high energy line breaks:

1. Auxiliary Steam System piping that is normally isolated
2. Condensate System piping that is normally isolated
3. Main Feedwater System piping that is normally isolated
4. 'F' Extraction Steam to 'F' Heaters, including the associated heater vents and drains
5. 'E' Extraction Steam to 'E' Heaters, including the associated heater vents and steam drains
6. 'E' Heater Drains to the 'E' Heater Drain Pumps

7. Portions of the High Pressure Injection System
    - RCP Seal Return piping
    - Letdown piping from the block orifice to the LDST
    - HPI Pump Suction piping from the LDST
    - HPI Pump C Discharge piping
8. Main Steam System piping that is normally isolated
9. Portions of the Moisture Separator Reheater Drain System
    - Outlet from the Moisture Separator Drain Pump Demineralizer Heat Exchanger
    - Piping that is normally isolated
10. Plant Heating System piping that is normally isolated
11. Portions of the Steam Seal Header that operate at or below atmospheric pressure
12. Low Pressure Injection System
13. Reactor Building Spray System
14. Emergency Feedwater System piping
15. Standby Shutdown Facility Auxiliary Service Water System piping
16. Steam Generator Blowdown piping
17. Nitrogen System piping
18. Electro-Hydraulic Control (EHC) System piping (oil system).

RAI 22 [H]

In the response to RAI 12(a), the licensee discussed the new term "Initial Operating Conditions." According to the discussion, this term is consistent with the analysis presented in Section 3.0 of the Mechanical Design Section (MDS) Report OS-73-2."

a) Please provide specific details, including a comparison of the Normal Plant Conditions, defined in Auxiliary Systems Branch (ASB) 3-1 and the Operation Modes defined in HELB report, to demonstrate that the statement is correct and applicable to this situation.

b) In the licensee's response to RAI 12(b), reference is again made to their response to RAI 3. As stated by the NRC staff in RAI 3, a complete list of the systems with a data summary on each should be assembled and made available to reviewers.

Duke Energy Response

The term “Initial Operating Conditions” is defined on Page 1-14 of the HELB Report (ONDS-351). The purpose of having the term “Initial Operating Conditions” is to establish a starting point, which defines the physical configuration of the plant and the plant conditions, when the HELB is postulated to occur. This is done, so that the pathways to a Safe Shutdown condition from the most severe initial conditions can be determined. The only relationship to the plant operating modes is that the HELBs are postulated to occur with the unit in Mode 1(Power Operation) at the 100% rated thermal power condition. The term is consistent with Section 3.0 of the MDS report OS-73-2 (ONDS-351 Reference 10.3.1) in that both documents identify the unit operating at 100% rated thermal power (Full Power) at the time the HELB is postulated to occur. The definition of “Initial Operating Conditions (or Normal Operating Conditions)” is not the same definition as “Normal Plant Conditions” defined on Page 1-15 of the HELB Report.

The term “Normal Plant Conditions,” is defined on Page 1-15 of the HELB Report and is based upon the definition of the same name provided on Page 3.6.1-16 of SRP 3.6.1 (ASB 3-1). The

ONS Operational Modes (taken from Table 1.1-1 of the ONS Technical Specifications) are defined on Page 1-14 of the HELB Report. A comparison of terms and ONS Operational Modes is provided below:

ONS Operational Modes (HELB Report Page 1-14)	HELB Report "Normal Plant Conditions" Description (HELB Report Page 1-15)	Definition of "Normal Plant Conditions" SRP 3.6.1 (Page 3.6.1-16)
1	Power Operation	Operation at Power
2	Startup	Reactor Startup
3	Hot Standby	Hot Standby
4	Hot Shutdown	Reactor Cooldown to Cold Shutdown Condition

For the second paragraph of this RAI please refer to the response to RAI 23b.

RAI 23 [H]

a) The licensee states that "Those systems that could be operated at high energy conditions for short periods of time during modes 1 through 4 were excluded." Please provide specific details on the criteria used to qualify "short periods of time," and provide the data that served as the bases for the calculations performed to demonstrate that the stated exclusions are justified.

b) In the licensee's response to RAI 13, reference is again made to their response to RAI 3. As stated by the NRC's staff in RAI 3, a complete list of the systems with a data summary on each should be assembled and made available.

Duke Energy Response

(a) As described in ONDS-351, a number of criteria were employed in the development of the high energy systems in which break locations would be postulated.

1. High energy systems were selected based on operating conditions existing at 100% of rated full power. The high energy boundaries for these systems were selected based on the normal operating configuration of the systems at 100% of rated full power. Piping downstream of a normally closed high energy boundary valve was excluded from high energy break consideration. "Short periods of time" were not applied to these high energy line break exclusions.
2. For systems not normally in operation, "short periods of time" were used to exclude these systems from high energy line break consideration.
   - Pipe breaks and critical cracks are not postulated on high energy lines that operate at high energy conditions less than 1% of the plant operating time during Modes 1 through 4.
   - Pipe breaks and critical cracks are not postulated on high energy lines that operate at high energy conditions less than 2% of the total system operating time.

The details were provided for the systems excluded using "short periods of time during Modes 1 through 4" in the response to RAI 14 of this letter.

b) A complete listing of systems that were excluded from high energy line break consideration was provided in response to RAI 21 of this letter. Provided below is a discussion of the exclusions that were made in accordance with the criteria provided in "a" above.

The Auxiliary Steam System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves AS-7, AS-311, (1)(2)(3) AS-34, (1)(2)(3)AS-40, (1)(2)(3)AS-26, 1AS-465, AS-22, 3AS-22, and 3AS-364. A number of these valves (AS-26, AS-465, AS-22, and AS-364) are manually operated valves that are normally closed with no planned actions to open them. AS-7 and AS-311 are also manually operated valves that are normally closed, but they are opened whenever the Auxiliary Boiler is placed into service. AS-34 is a motor-operated valve that is normally closed, but it would be opened during a unit startup or shutdown to provide steam to the 'E' Feedwater Heaters to maintain sufficient feedwater temperatures. AS-40 is also a motor-operated valve that is normally closed, but the valve is opened during startup and shutdown to supply steam to the condenser steam air ejectors (CSAEs) to maintain condenser vacuum. The use of "short periods of time" was not applied to these normally isolated lines.

The Condensate System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves (1)(2)(3)C-425, (1)(2)(3)C-426, (1)(2)(3)C-427, (1)(2)(3)C-124, (1)(2)(3)C-98, (1)(2)(3)C-99, (1)(2)(3)C-311, (1)(2)(3)C-314, (1)(2)(3)C-320, and (1)(2)(3)C-321. A number of these valves (C-98, C-99, C-320 and C-321) are manually operated valves that are normally closed with no planned actions to open them. The condensate booster pump (CBP) minimum recirculation valves (C-425, C-426, and C-427) automatically open (on the operating pump) when the total condensate flow decreases to 4500 gpm or below. The feedwater seal injection pump inlet valves (C-311 and C-314) are normally in automatic. The valves will automatically open when the associated seal injection pump receives a start signal due to low differential pressure between the CBP discharge pressure and the main feedwater pump suction pressure. The condensate recirculation path to the Upper Surge Tank (UST) is normally isolated by a closed motor-operated valve (C-124). The valve may be opened during unit startup or shutdown. The use of "short periods of time" was not applied to these normally isolated lines.

The Main Feedwater System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves (1)(2)(3)FDW-53, (1)(2)(3)FDW-65, (1)(2)(3)FDW-38, (1)(2)(3)FDW-47, (1)(2)(3)FDW-74, (1)(2)(3)FDW-76, (1)(2)(3)FDW-200, (1)(2)(3)FDW-262, (1)(2)(3)FDW-263, (1)(2)FDW-279, (1)(2)(3)FDW-280, (1)(2)(3)FDW-283. A number of these valves (FDW-262, FDW-263, FDW-279, FDW-280, and FDW-283) are manually operated valves that are normally closed. The valves may be opened when draining the feedwater system. The main feedwater pump minimum recirculation valves (FDW-53 and FDW-65) are normally in automatic. While in the automatic mode, the valves throttle open when the applicable main feedwater pump suction flow decreases to about 2300 gpm. These valves may be operated in the manual mode during startup and shutdown evolutions to maintain total condensate flow within a prescribed flow band. The main feedwater cleanup line is normally isolated from the main feedwater system by closed motor-operated valves (FDW-74, FDW-76, and FDW-200). The valves may be opened during unit startup and shutdown when feedwater cleanup is desired. Each main feedwater header is equipped with a line that directs flow to the auxiliary nozzles of the associated steam generator. Each of these lines are normally isolated from the high energy portion of the main feedwater system by a closed motor-operated valve (FDW-38 and FDW-47). The valves are equipped with an automatic signal to open the valves on a loss of all four reactor coolant pumps or a loss of both main feedwater pumps. In addition, the valves may be opened during startup and shutdown evolutions. The use of "short periods of time" was not applied to these normally isolated lines.

The High Pressure Injection (HPI) System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves (1)(2)(3)HP-116 and

(1)(2)(3)HP-409. The normally isolated portion of the HPI system can either be pressurized during certain accident conditions or for routine performance testing. The details of the historical review that was performed to quantify the "short periods of time" while subjected to high energy conditions is provided in the response to RAI 14 of this letter

The Main Steam System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves (1)(2)(3)MS-153, (1)(2)(3)MS-155, (1)(2)(3)MS-19, (1)(2)(3)MS-22, (1)(2)(3)MS-28, (1)(2)(3)MS-31, (1)(2)(3)MS-37, and (1)(2)(3)MS-38. A number of these valves (MS-153, MS-155, MS-37, and MS-38) are manually operated valves that are normally closed. The atmospheric dump block valves (MS-153 and MS-155) may be opened for testing during startup or shutdown. In addition, the valves may be opened following events where the turbine bypass valves are not available. There are four turbine bypass valves (MS-19, MS-22, MS-28, and MS-31) that are normally closed. The discharge of each turbine bypass valve is connected to a common discharge header. The common discharge header is then divided into three lines that are directed to the main condenser (one line per condenser). During normal operation, these lines are subjected to vacuum conditions. Following a main turbine trip or planned shutdown of the main turbine, the turbine bypass valves (TBVs) open as necessary to control main steam pressure at the desired setpoint. The TBVs are utilized to cool the Reactor Coolant System (RCS) down to LPI entry conditions. During startup evolutions, the TBVs are initially opened to pull a vacuum on the steam generators. Once RCS heatup is commenced, the TBVs would be closed to allow the heatup to continue. The TBVs may be throttled open during periods of startup where the heatup process is placed on hold. The TBVs are also throttled open during reactor power increases until the main turbine is placed on -line. The use of "short periods of time" was not applied to these normally isolated lines. However, statements were made in sections 4.1.8, 5.1.8, and 6.1.8 of ONDS-351 that describe the downstream piping from the TBVs as not being pressurized more than 2% of the operating time of the main steam system. These statements will be corrected.

The Moisture Separator Reheater Drain (MSRD) System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves (1)(2)(3)HD-25, (1)(2)(3)HD-26, (1)(2)(3)HD-27, (1)(2)(3)HD-28, (1)(2)(3)HD-29, (1)(2)(3)HD-30, (1)(2)(3)HD-102, (1)(2)(3)HD-103, (1)(2)(3)HD-41, (1)(2)(3)HD-43, (1)(2)(3)HD-56, (1)(2)(3)HD-58, 3HD-70, 3HD-72, 3HD-85, 3HD-87, 3HD-453, 3HD-454, 3HD-541, and 3HD-540. A number of these valves (HD-41, HD-43, HD-56, HD-58, 3HD-70, 3HD-72, 3HD-85, 3HD-87, 3HD-453, 3HD-454, 3HD-541, and 3HD-540) are manually operated valves that are normally closed. Drain valves (3HD-453, 3HD-454, 3HD-540, and 3HD-541) may be opened to drain the system. HD-102 and HD-103 are motor-operated valves that are normally closed. The second stage reheater (SSRH) drain tank dump valves (HD-25 and HD-26) are normally closed at full power. The valves will open when the main turbine is shutdown or if the normal level control valves (HD-92 and HD-95) are not capable of maintaining proper water level. The first stage reheater (FSRH) drain tank dump valves (HD-29 and HD-30) are normally closed at full power. The valves will open when the main turbine is shutdown or if the normal level control valves (HD-66 and HD-81) are not capable of maintaining proper water level. The moisture separator reheater (MSRH) drain tank dump valves (HD-27 and HD-28) may be throttled open at full power, depending on the mode of operation selected. If the MSRH drain tanks are being fed forward, then the dump valves would be closed. However, the dump valves discharge to the condenser hotwell, which is normally under vacuum conditions. In fact, all of the dump valves discharge to the condenser hotwell. The use of "short periods of time" was not applied to these normally isolated lines.

The Plant Heating System is a high energy system that is normally in service. High energy boundaries were established at normally closed valves (1)(2)(3)AS-182, (1)(2)(3)AS-70, (1)(2)(3)AS-75, (1)(2)(3)AS-73, (1)(2)(3)AS-78, 2AS-108, and PH-123. All of these valves are manually operated. Only AS-182 is repositioned by procedure. This valve is opened on the applicable unit when it is desired to place the Reactor Building Purge system in service and the outside air temperature is below 60°F. The applicable unit is below Mode 4 when this alignment occurs. The use of "short periods of time" was not applied to these normally isolated lines.

The Low Pressure Injection (LPI) System does not normally operate in a high energy condition. This system is normally isolated from the RCS by closed motor-operated valves. The system is charged from the BWST by two normally open motor-operated valves. The system is normally pressurized by the head of the BWST. Both the pressure from the BWST and the temperature in the system are below the threshold for high energy conditions. The LPI pumps are routinely tested during power operation. The testing does not subject the LPI system piping to high energy conditions. During the latter stages of plant cooldown, the system is isolated from the BWST and aligned to the RCS by opening the normally closed motor-operated valves from the RCS. The RCS is aligned to the LPI system after RCS pressure has been reduced to below 300 psig and RCS temperature has been reduced below 246 degrees F. This subjects the LPI system to high energy conditions until the RCS is cooled to 200 degrees F (or below) and depressurized to 275 psig (or below). Likewise, during the initial stages of RCS heatup and pressurization for unit startup activities, the LPI system is aligned to the RCS where conditions subject the LPI system to high energy conditions. The total time the LPI spends in high energy conditions is typically short in duration. A historical review was performed for startup/shutdown evolutions on all three units using OAC data to quantify the "short periods of time" while subjected to high energy conditions. The historical review period for Unit 1 was from 7/8/1999 to 6/1/2008. The historical review period for Unit 2 was from 12/16/1999 to 12/12/2008. The historical review period for Unit 3 was from 5/21/2000 to 12/18/2007. The LPI system experienced high energy conditions for approximately 32 [24 hour] days on Unit 1, approximately 17 days on Unit 2, and approximately 14 days on Unit 3 for the time period reviewed

The Reactor Building Spray, Emergency Feedwater, and the Standby Shutdown Facility Auxiliary Service Water systems are not normally in operation. These systems are either operated during certain accident conditions or for routine performance testing. The details of the historical review that was performed to quantify the "short periods of time" while subjected to high energy conditions is provided in the response to RAI 14 of this letter.

The Steam Generator Blowdown System is not normally in service. Each steam generator is equipped with a blowdown line that directs flow to the main condenser. Both of the blowdown lines are normally isolated from the high energy portion of the steam generators by closed manually operated valves located inside the reactor building. During unit startup, it is desired to establish steam generator blowdown to control the water chemistry inside the steam generators. The total time the steam generator blowdown piping spends in high energy conditions is typically short in duration. A historical review was performed for startup and shutdown evolutions on all three units using OAC data to quantify the "short periods of time" while subjected to high energy conditions. The historical review period for Unit 1 was from 7/8/1999 to 6/1/2008. The historical review period for Unit 2 was from 12/16/1999 to 12/12/2008. The historical review period for Unit 3 was from 5/21/2000 to 12/18/2007.The SG Hot Blowdown piping experienced high energy conditions for approximately 35 days on Unit 1, approximately 32 days on Unit 2, and approximately 35 days on Unit 3 for the time period reviewed.

RAI 24 [H]

In the response to RAI 14, in the HELB report, the licensee states "the subject piping....is seismically designed, analyzed, and supported... to assure that the Class G/F boundary is seismically protected." Please provide references to the appropriate calculations and evaluations or assessments that demonstrate the validity of the statements that are made.

Duke Energy Response

The piping stress analysis of the Main Feedwater system is contained in the following calculations OSC-336 (Unit 1), OSC-454 (Unit 2), and OSC-512 (Unit 3). These calculations document the seismically designed 'overlap boundaries' that extend beyond the Class G/F boundary.

RAI 25 [T/H]

In RAI 15, the NRC staff requested a list of all the equipment types (including manufacturer and model number) that need to be qualified for the environmental conditions of the LAR, with confirmation that all the identified components are qualified in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.49.

The licensee's response included a tabulated list with a statement referring to the list:

All of the above components existed prior to the LAR and that none of these components needed to be added to the Equipment Qualification program as a result of the LAR. All PSW [Protected Service Water] System components will be qualified for the applicable environment. None of the above listed components were replaced as a result of the LAR. The temperature and pressure profile for components located inside the [East Penetration Room] EPR [and] & [West Penetration Room] WPR was changed as a result of a new analysis for the postulated breaks on the main steam piping and main feedwater piping located inside the East Penetration Room. The above components were reviewed for the new pressure and temperature profiles and found to be qualified. The component evaluations are documented in calculation [Oconee site calculation] OSC-8505 [Oconee Nuclear Design Study] (ONDS-351, Revision 2, Ref. 10.2.17).

Define what the protected service water (PSW) components are and state when and how the PSW system components will be qualified for the applicable environment.

Duke Energy Response

A list of the engineered components for the Protected Service Water System (PSW) is provided below and type of environment (Harsh/Mild or N/A) as defined by Duke Nuclear System Directive NSD-303 (Environmental Qualification Program) and the Oconee Environmental Qualification Criteria Manual. A Mild or N/A designation indicates the component does not fall within the scope of 10CFR50.49.

The following is a summary of the processes used to ensure that the PSW system components will be qualified for the applicable environment.

The PSW engineered components are procured using Duke Engineering Directives Manual EDM-140 (Procurement Specifications For Equipment). The procurement documents are in the form of a Procurement Specification or a Technical Requirements Document (TRD).

A Procurement Specification is a controlled document that defines the requirements for the procurement of the equipment. A TRD is a "mini-specification" that provides the key technical and administrative information necessary to communicate the requirements to the equipment supplier.

Both the Procurement Specification and TRD contain equipment functional requirements for normal system operating conditions including temperature, pressure, flows, voltage, current, frequency, etc., seismic loads and for accident environmental conditions including temperature, pressure, humidity and radiation.

For equipment requiring Environmental Qualification (EQ), the procurement documents contain the performance requirements and qualification criteria and environmental parameters. The supplier submits a Qualification Test Plan which is reviewed by Duke to ensure that the proposed plan meets or exceeds the equipment specifications. Upon completion of the environmental testing, the supplier submits a Qualification Test Report which is processed in accordance with NSD-303.

If applicable, the equipment is identified as EQ-related in the Equipment Database (EDB) and included in the EQ Maintenance Manual (EQMM). The EDB identifies the EQMM section and qualified life for the equipment. The EQMM contains the information needed for maintaining the equipment environmental qualification including maintenance and replacement requirements.

PSW Component	Environment Type
PSW Primary Pump	Mild
PSW Booster Pump	Mild
PSW Building QA-1 HVAC System	Mild
Auxiliary Building PSW Pump Room Safety-Related Ventilation System	Mild
Motor Operated Throttle Valve (DMV-1710) for HP-31 Bypass Line	Harsh
Motor Operated Throttle Valves (DMV-1464 and DMV-1471)	Harsh
Modulating Solenoid Operated Throttle Valves (DMV-1463)	Mild (Modulation and Position Equipment) Harsh (Valve and Operator)
15 and 45 KVA QA-1 600/208/120 VAC Transformers	Mild
QA-1 Medium Voltage Unit Substation and Manual Disconnect Switch	Mild
QA-1 600 VAC Load Center and 5 MVA Transformer)	Mild
QA-1 125 VDC Distribution Center and Power Distribution Panels	Mild
QA-1 600 VAC Motor Control Centers	Mild
QA-1 208/120 VAC Power Distribution Panels	Mild
QA-1 200 HP Booster Pump Motor	Mild
QA-1 2000 HP Primary Pump Motor	Mild
QA-1 125 VDC Batteries	Mild
QA-1 125 VDC Battery Chargers	Mild
QA-1 5 kV Motor Operated Manual Transfer Switches for HPI Pumps	Harsh
QA-1 5 kV Manual Disconnect/Alignment Switches for HPI Pumps	Harsh
QA-1 600 VAC Manual Transfer Switches	Mild
QA-1 600 VAC Automatic Transfer Switches	Mild
QA-1 600 and 120 VAC Manual Transfer Switches	Mild
Keowee QA-1 Medium Voltage Switchgear, Protective Relay Board and Non-QA-1 Electrical Support Equipment	Mild

PSW Component	Environment Type
Keowee Isolated Bus Junction Box	Mild
Non-Safety PSW Building Air Conditioning Equipment	N/A
Isolation Relay Panels	Mild
Motor Operated Valve DMV-1711- HP-31 Outlet Isolation Valve.	Harsh
Handwheel Valve DHV-1455 - PSW Test Line Isolation Valve	N/A
Chainwheel PSW Isolation Valves DHV-1459 and DHV-1488	N/A
Motor Operated Valve DMV-1462 SG Common Header Isolation Valve	Harsh
Check Valves DMV-1472 and DMV-1483	N/A
Chainwheel Isolation Valve DMV-1473	N/A
Chainwheel Isolation Valve DMV-1478	N/A
Chainwheel Isolation Valves DMV-1484, DMV-1485 and DMV-1486	N/A
Handwheel Test Valve DMV-1487	N/A
Handwheel Valve DMV-1456 Auxiliary Service Water Suction Valve	N/A
Bargraph Indicators for 1HPI P-0025, 2HPI P-0152 and 3HPI P-0152	Mild
SBM Switches for 1RC CS-155/156, 2RC CS-155/156, 3RC CS-155/156, 1RC CS-157/158, 2RC CS-157/158, 3RC CS-157/158, 1RC CS-159/160, 2RC CS 159/160, 3RC CS-159/160, 1HPI CS-0024, 2HPI CS-0024, 3HPI CS-0024, 1HPI CS-0026, 2HPI CS-0026, 3HPI CS-0026, 1HPI CS-PUA, 2HPI CS-PUA, 3HPI CS-PUA, 1HPI CS-PUB, 2HPI CS-PUB, 3HPI CS-PUB, 1HPI CS-PUMP, 2HPI CS-PUMP, 3HPI CS-PUMP, 0PSW CS-0001, 0PSW CS-0002 and 0PSW CS-0003	Mild
SBM Switches and ET-16 Indicating lights for 1PSW CS-1X1, 1PSW CS-1XK, 2PSW CS-2XH, 2PSW-2XI, 2PSW CS-2XK, 3PSW CS-3XH, 3PSW CS-3XI, 3PSW CS-3XK, 1PSW LI 1XIG, 1PSW LI-1XIGI, 1PSW LI-1XIR, 1PSW LI-1XIR1, 1PSW LI-1XKG, 1PSW LI-1XKG1, 1PSW LI-1XKR, 1PSW LI-1XKR1, 2PSW LI-2XHG, 2PSW LI-2XHG1, 2PSW LI-2XHR, 2SPW LI-2XHR1, 2PSW LI-2XG, 2PSW LI-2XIG1, 2PSW LI-2XIR, 2PSW LI-2XIR1, 2PSW LI-2XKG, 2PSW LI-2XKG1, 2PSW LI-2XKR, 2PSW LI-2XKR1, 3PSW LI-3XHG, 3PSW LI-3XHG1, 3PSW LI-3XHR, 3PSW LI-3XHR1, 3PSW LI-3XIG, 3PSW LI-3XIG1, 3PSW LI-3XIR, 3PSW LI-3XIR1, 3PSW LI-3XKG, 3PSW LI-3XKG1, 3PSW LI-3XKR and 3PSW LI-3XKR1	Mild
PSW Building Fire Detection System	N/A
PSW Remote Motor Starter for SSF Submersible Pump	Mild

RAI 26 [H]

a) Please provide the calculations discussed in the response for review: OSC-8505 (ONDS-351, Revision 2, Reference 10.2.17), and OSC-8104 (ONDS-351, Revision 2, Reference 10.2.3).

b) Also, in the response to RAI 16, the licensee states that: "For postulated HELBs in other areas of the Auxiliary Building, equipment qualification is not required. Either the loss of any shutdown components in these areas would not preclude achieving and maintaining a safe shutdown condition, or adverse environmental conditions are not generated...."

Please provide reference to the documentation that forms the basis of this evaluation and conclusion.

Duke Energy Response

a.) The current revisions of both calculations, OSC-8104 & OSC-8505, have been electronically scanned and uploaded to the SharePoint site for NRC Staff review.

b.) The results of the interaction analyses of the postulated HELBs in the Auxiliary Building of each Oconee Nuclear Station are documented in Sections 4.2.1 (Unit 1), 5.2.1 (Unit 2), and 6.2.1 (Unit 3) of the Oconee HELB Report (ONDS-351). These evaluations are based upon the determination of the High Energy piping and break locations documented in calculations OSC-7516.01, Unit 1 (ONDS-351 Reference 10.2.2); OSC-7517.01, Unit 2 (ONDS-351, Reference 10.2.39); and OSC-7518.01, Unit 3 (ONDS-351 Reference 10.2.52). These HELB locations are listed in the respective Tables 4.1, 5.1, & 6.1 in the HELB Report. The plant equipment required for attainment and maintenance of the Safe Shutdown condition and subsequent unit cool down to the Cold Shutdown Condition are documented in calculations OSC-8089.01 & OSC-8089.02 (ONDS-351 References 10.2.4 & 10.2.15, respectively). All direct HELB interactions in any unit are documented in the respective Tables 4.2, 5.2 & 6.2 of the HELB Report. The indirect HELB interactions are described in the interaction analyses results in the HELB Report. The statement generated for RAI 26 is the overall summary statement for those areas of the Auxiliary Buildings beyond the East and West Penetration rooms, based upon the described interaction analyses in the HELB Report.

RAI 27 [T/H]

In RAI 17, the licensee states, "The primary and backup cables associated with the 125 vdc vital [instrumentation and control] I&C system will be rerouted out of the Turbine Building to eliminate vulnerabilities to HELB and/or Tornado events."

Please provide a reference to the documentation that validates this statement. Note that there is no commitment to reroute the cables in the commitments listed in the HELB report in the Unit 3 LAR.

Duke Energy Response

The rerouting of the primary and backup cables associated with 125 VDC Vital I&C System for each ONS Unit is part of the Protected Service Water (PSW) System modification for the Oconee Nuclear Station. The rerouting of these cables out of the Turbine Building is documented as one of the physical configuration changes for the PSW System in the Engineering Change Request for the PSW System.

Because these cable reroutes are contained within the PSW System modification scope and would be required in order to meet the PSW design criteria, a separate commitment is not necessary. Moreover, the description of the cable reroute modification on Page 9-3 of the HELB Report (ONDS-351, Rev.2) identifies this modification as part of the PSW project, which is a commitment. This modification was added to Section 9.0 of the HELB Report as a separate item in order to emphasize its importance for future reference.

RAI 28 [H]

a) It is not clear how the response to RAI 18 confirms that appropriate drainage is provided for all junction boxes. How were the holes sized?

b) Please provide details of, or reference to, an inspection plan or a controlled document that specifies the frequency of weep hole inspections.

c) Please address whether it is possible that the weep holes could provide a pathway for water to enter the enclosures?

Duke Energy Response

a) The outside-containment Viking electrical penetration enclosures will have three (3) 1/4 inch diameter weep holes located at the left, center and right of the bottom of the enclosure. The weep hole sizing and location were determined using Duke procedural guidance and engineering judgment based on the Viking enclosure design.

b) Inspection of electrical penetrations enclosure weep holes is included in procedure IP/0/A/3010/011 (Inspection of Electrical Penetration Enclosures). This procedure (including weep hole inspection) is performed on a six year frequency under an ONS model Work Order.

c) Due to evidence of past penetration room roof leaks resulting in water entry into the Viking enclosures, the weep holes were installed as an enhancement to the Viking electrical penetration enclosure design. The decision to install the weep holes is also consistent with guidance found in USNRC Information Notice 89-63. The Viking enclosures are not environmentally sealed. Postulated water entry through the weep holes is not a concern since the normal and accident conditions for the penetration rooms does not include spray or submergence.

RAI 29 [H]

The licensee's response to RAI 19 does not address RAI 19, as it does not discuss how the licensee ensured that failure of nonsafety-related components would not adversely affect the safety function of a safety-related component under postulated environmental conditions. Please address this issue.

Duke Energy Response

The Environmental Qualification (EQ) Program for the Oconee Nuclear Station (ONS) is governed by Duke Energy's Nuclear System Directive 303 (NSD-303), which is referenced in Section 3.11, Environmental Design of Mechanical and Electrical Equipment, of the ONS UFSAR.

NSD-303 identifies which electrical components are within the scope of 10CFR 50.49 and require environmental qualification. For non safety-related electrical equipment the directive requires environmental qualification of:

"Non QA Condition 1 (non safety-related) electrical equipment located in a postulated harsh environment, whose failure could prevent a safety function or mislead the plant operator."

Section 303.2.4, Evaluating Non-Safety Equipment, of NSD-303 provides additional guidance for determining if non safety-related components require EQ.

The method used by Oconee for identification of non safety-related electric equipment, whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions or mislead the operator is as follows:

1. A list was generated of safety-related electrical equipment as defined in paragraph (b)(1) of 10CFR 50.49 that are required to remain functional during or following design-basis Loss of Coolant Accident (LOCA) or High Energy Line Break (HELB) Accidents. The LOCA/HELB accidents are the only accidents that result in significantly adverse environments to electrical equipment, which is required for safe shutdown or accident mitigation. The list was based on reviews of the ONS, Units 1, 2, and 3 UFSAR and the elementary diagrams, connection diagrams, P&ID's (flow diagrams), and cable lists.
2. The elementary wiring diagrams of the safety-related electrical equipment identified in Step 1 are reviewed to identify any (non safety-related) auxiliary devices electrically connected directly into the control circuitry or power circuitry of the safety-related equipment, whose failure due to postulated environmental conditions could prevent the required operation of the safety-related equipment; and
3. In reviewing the environmental qualification documentation for class 1E equipment, the function of the equipment is reviewed via P&IDs, component technical manuals, and/or systems in the UFSAR. Any directly connected mechanical auxiliary systems to electrical equipment, which are necessary for the safety-related electrical equipment to perform its safety function are considered in the qualification of the class 1E equipment.
4. Non safety-related electrical circuits indirectly associated with the safety-related electrical equipment identified in Step 1 by common power supply or physical proximity are considered by a review of the ONS electrical design including the use of applicable industry standards (e. g. IEEE) and the use of properly coordinated relays, contacts, isolation amplifiers, individual output relays, circuit breakers, and fuses for electrical fault protection.
5. For those identified non safety-related components in a harsh environment that could adversely affect safety-related components or misled the plant operator identified in Steps 2-4, environmental qualification criteria are established, and the EQ of the component is documented per the requirements of NSD-303.

This methodology has been approved for ONS by NRC letter dated March 20, 1985. This letter has been uploaded to the SharePoint site.

RAI 30 [H]

In the response to RAI 21, the licensee states:

The postulated HELBs in the Auxiliary Building do not directly or indirectly impact any of the remaining components or their support systems. Cables for these components that are routed through the East or West Penetration Rooms are qualified for the adverse environmental conditions in these rooms, created by these postulated Main Feedwater or Main Steam line HELBs. The qualification of the cables is documented in calculation OSC-8505 (ONDS-351, Rev. 2, Ref. 10.2.17).

The NRC staff requests a copy of calculation OSC-8505 for review.

Duke Energy Response

The calculation has been electronically scanned and uploaded to the SharePoint site for Staff review.

RAI 31 [H]

NUREG 0800, Section 3.6.2, Item I (OL) 4, Revision 1, and NUREG 0800, Section 3.6.2, Item 1.7, Revision 2, identify as a specific area of review:

The design adequacy of systems, components, and component supports to ensure that the intended design functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe-whip or jet impingement loadings.

In order to cover this area of review adequately, the following calculations are requested for review:

1. Calculation OSC-7516.01, "ONS Unit 1 HELB High Energy Line Break Stress Evaluation" (Application of criteria used to define break and crack locations and configurations).
2. OSC-7516.03, "Unit I HELB Turbine Building Structural Interactions Evaluations" (Application of methodology used to define the forcing functions, including the jet thrust reaction at the postulated pipe break or crack location and jet impingement loadings on adjacent safety-related, structures, systems and components (SSCs).
3. Calculation OSC-7516.04, "Safe Shutdown Equipment Damage Assessment for HELB -ONS Unit 1" (Application of criteria used to define break and crack locations and configurations).
4. OSC-8505, "Oconee HELB EO Analysis for Penetration Rooms" (Application of methods used to validate the equipment qualification of the Shutdown Electrical Components).

Duke Energy Response

The current licensing basis for the Oconee Station for High Energy Line Breaks is the Giambusso/ Schwencer letters. Oconee is not licensed to the Standard Review Plan (SRP) 3.6.2 and does not propose to be licensed as such in the future.

In order to fully evaluate the Oconee High Energy Line Break program per the statement quoted above the following calculations are required to be reviewed:

- OSC-7516.01
- OSC-7516.02
- OSC-7516.03
- OSC-7516.04
- OSC-8556
- OSC-7516.07
- OSC-7516.08
- OSC-7516.09
- OSC-7516.10

The purpose of each of these calculations is provided on Pages 1-2 to 1-4 & 1-9 of the HELB Report. All of these calculations are available in PDF form except for OSC-7516.03. Calculation OSC-8505 is also available in PDF form. All of the calculations except for OSC-7516.03 have been electronically scanned and uploaded to the SharePoint site for NRC Staff review.

In order to fully examine most of these calculations, review from the hard copies of them is strongly recommended. Examination of these calculations on the computer network is often awkward in orientation, the resolution of the drawings is not, in general, acceptable for review, and OSC-7516.03 could not be scanned. All of these calculations are readily available at the Oconee site and can be made immediately available for NRC review.

RAI 32 [H]

The HELB report in the Unit 3 LAR refers to ONS "HELB Outside Containment Walkdown Criteria & Requirements," (Reference 10.3.17). Please provide the document for review.

Duke Energy Response

Revision 3 of the "HELB Outside Containment Walkdown Criteria & Requirements" has been electronically scanned and uploaded to the SharePoint site for NRC Staff review.

RAI 33 [H]

Section 2.1 of the HELB report in the Unit 3 LAR states that:

> *The High Energy (Piping) lines are those lines that during Initial Operating Conditions the fluid inside of the pipe has either or both of the following conditions:*
>
> 1. *A normal operating temperature greater than 200 ^{0}F.*
> 2. *A normal operating pressure greater than 275 psig [pounds per square inch gage].*

According to the NUREG-0800 definition, a piping system with either a 200 of service temperature or a 275 psig design pressure would be considered high energy piping, while under the definition in the Unit 3 LAR they would not. Is there any piping that was eliminated from the high energy piping population because it had an exact 200 of service temperature?

Duke Energy Response

A review of the piping systems evaluated in calculation OSC-8385 (HELB Report Reference 10.2.1) showed that no piping has been eliminated from the high energy piping population based only upon its operating temperature, and there is no piping listed in the calculation as operating at 200°F. Moreover, all piping in the calculation with documented operating temperatures above 200°F is classified as high energy. The high energy piping determinations made beyond this calculation did not use the temperature criterion to eliminate piping from the high energy piping population.

RAI 34 [H]

Section 2.2.2 of the HELB report in the Unit 3 LAR states the following:

> *For branch connections to piping runs, a branch, appropriately modeled in a rigorous stress analysis with run flexibility and applied branch line movements included and where the branch connection stress is accurately known, will use the stress criteria for seismically analyzed piping lines; for postulating HELB locations.*

Do the "applied branch line movements" include amplified seismic response of the run piping at the branch line termination? If not, please address how "stress is accurately known" at that location?

Duke Energy Response

The applied branch line movements include both thermal and amplified seismic movements from the run pipe. The intersection of the run and branch line is represented as an anchor point in the branch line piping analysis model. The run pipe movements are applied as anchor motion(s) to the branch line piping analysis model.

RAI 35 [H]

The HELB report in the Unit 3 LAR states:

2.2.3 High Energy Break Type Criteria

The following criteria are used to identify the [high energy] HE break types, required to be postulated at the identified break locations in the ONS. There are three (3) types of HELBs at the ONS. These include circumferential breaks, longitudinal breaks, and critical cracks. The definition and description of each of these break types are provided in Section 1.5. The criteria for each break type are as follows (References 10.1.1, 10.1.3, & 10.3.17)[1]:

1. Circumferential Breaks are to be postulated in HE lines that exceed 1 inch in nominal pipe size.
2. Longitudinal Breaks are to be postulated in HE piping that has a nominal pipe size of four (4) inches or greater.
3. Critical Cracks are to be postulated on seismically analyzed HE piping that exceeds 1 inch in nominal pipe size (See Section 2.2.2 for exceptions).
4. HELBs of any type are not postulated on HE piping that has a nominal pipe of 1 inch or less.
5. Only circumferential breaks are to be postulated at terminal ends of HE piping runs. (Longitudinal breaks are not postulated at terminal ends.)
6. Longitudinal breaks are to be postulated only at intermediate break locations on HE piping runs.
7. For piping that has a nominal pipe size of four (4) inches or greater both circumferential and longitudinal breaks are to be postulated at the intermediate break locations but not concurrently.
8. Longitudinal breaks are to be postulated parallel to the pipe axis and orientated at any point on the pipe circumference.
9. The break area of a longitudinal break is equal to the effective cross-sectional flow area of the pipe immediately upstream of the break location.
10. Longitudinal breaks are not required to be postulated at branch connections.

Criterion 1 states that circumferential breaks are to be postulated in HE lines that exceed 1 inch in nominal pipe size. This concurs with 3.(b) of BTP ASB 3-1 (Appendix B of SRP 3.6.1, Revision 1, July 1981).

Provide documentation that the cited stress range condition is not exceeded in any lines that are 1 inch or smaller nominal pipe size.

Criterion 2 states that longitudinal breaks are to be postulated in HE piping that has a nominal pipe size of four (4) inches or greater. This concurs with 3.(a) of BTP ASB 3-1 (Appendix B of SRP 3.6.1, Revision 1, July 1981). Provide documentation that the cited stress range condition is not exceeded in any lines that are smaller than 4 inches nominal pipe size.

For Criterion 3, please explain the significance of postulating critical cracks on "seismically analyzed" HE piping that exceeds 1 inch in nominal pipe size. Please address the difference between seismically analyzed piping versus piping analyzed for all necessary load cases.

Criterion 10 states longitudinal breaks are not required to be postulated at branch connections. The

[1] Note that bullets are used in the original text. Numbers are used here for clear reference below.

criterion appears to assume that branch connections are synonymous with terminal ends.

Provide the bases where, the branch runs are not classified as part of a main run in the stress analysis, that the branch runs does not have a significant effect on the main run behavior.

Duke Energy Response

Response to questions regarding ONDS-351, Section 2.2.3(1) and (2): The Giambusso/Schwencer criteria specifies in (3) "The criteria used to determine the pipe break orientation at the break locations as specified under (2) above should be equivalent to the following:

(a) Longitudinal breaks in piping in runs and branch runs, 4 inches nominal pipe size and larger, and/or
(b) Circumferential breaks in piping runs and branch runs exceeding 1 inch nominal pipe size."

BTP ASB 3-1 (Appendix B or SRP 3.6.1 Revision 1) sections 3(a) and 3(b) regarding stress ranges apply to Class 1 piping. There is no Class 1 piping outside containment at Oconee Nuclear Station. Since the ONDS evaluates the high energy line breaks outside containment, this provision of BTP ASB 3-1 does not apply.

Response to question regarding ONDS Section 2.2.3(3): Critical cracks were only postulated on rigorously analyzed piping that included seismic loading. There is no distinction between seismically analyzed piping versus piping analyzed for all necessary load cases. By definition, safety related equivalent Class 2 & 3 piping is analyzed for all necessary load cases at Oconee Nuclear Station. Section 2.2.2 of ONDS-351 addresses the applicable load cases for seismically analyzed piping. The fifth bullet item under Section 2.2.2 notes that for piping that is seismically analyzed the applicable load cases include internal pressure, dead weight (gravity), thermal, and seismic (OBE). Water-hammer load cases, as they applied to specific systems, were also included. The term, "seismically analyzed piping" was made to distinguish such piping from other piping systems at Oconee Nuclear Station that are analyzed, but are not analyzed for seismic loading. There are high energy piping systems at Oconee Nuclear Station that are rigorously analyzed for internal pressure, dead weight (gravity) and thermal, but not for seismic loading.

Response to question regarding ONDS Section 2.2.3(10): The intent of this criterion is that longitudinal breaks were not postulated for the branch pipe part of the branch connection. If the branch run piping was not classified as part of the main run in the stress analysis (e.g. the branch run was analyzed in a separate model), an anchor was assumed at the intersection of the branch run and main run. Anchor movements from the run piping were applied to the branch piping. A terminal end was assumed at the anchor location and a full circumferential break was postulated at the terminal end. The postulation of a circumferential break at the terminal end is equivalent to a longitudinal break of the run piping. For further discussion on analysis methodology of branch piping, see the response to RAI 18.

RAI 36 [T/H]

Section 3.4 of the HELB report in the Unit 3 LAR provides design criteria for the main steam isolation valves (MSIVs) to be installed. Are these valves to be designed to close against a flow resulting from an HELB immediately downstream?

Duke Energy Response

The planned modification to install the MSIVs includes a design requirement to close against the flow resulting from the postulated HELBs in the main steam line downstream of the MSIVs. Although the MSIVs have not yet been purchased, the procurement specification for the MSIVs has been released to vendors for quotes. The specification requires the valves to be capable of closing on demand when MS pressure reaches the closing setpoint (approximately 550 psig main steam pressure) with a flow rate corresponding to a double-ended break in the 36-inch (34-inch inside diameter) main steam line inside the turbine building. The flow rate specified for automatic closure is approximately 20,000,000 lbm/hr.

RAI 37 [T/H]

Provide the following information for the installation of the MSIVs.

1. Valve functional design, drawing, and specifications
2. Qualification program for demonstrating that the MSIV is capable of performing its specified functions under design basis conditions.
3. In-service testing program for monitoring and ensuring that the MSIV is capable of performing its functions under specified conditions.
4. Qualification test report.

Duke Energy Response

(1)(2)(3)(4) A procurement specification for the MSIVs (OSS-0245.00-00-0019) was developed and vendors have provided bids which are currently undergoing technical/commercial evaluation. The MSIV vendor has not yet been selected. The MSIV procurement specification has been posted on the SharePoint site (placed in the HELB RAI response documents folder). The valve drawings, qualification plans, in-service testing and qualification tests will not be available until the valve manufacturer has been selected and the procurement process started.

RAI 38 [T/H]

Describe the inadvertent closure of Main Steam Isolation Valves (MSIVs) and the affect on the plant.

Duke Energy Response

The planned modification to install the MSIVs includes performing the safety analysis for an inadvertent closure of a MSIV. The final design of the MSIVs and the associated calculations have not been completed.

RAI 39 [T/H]

For the operator actions associated with tornado and HELB mitigation provide the following information:

1. What is the required training?
2. How often will the training occur?
3. Will simulator training be used?

Duke Energy Response

Operators will receive initial classroom and simulator training on the Protected Service Water (PSW) modifications and the associated procedures that will be used by the operators to place the system(s) in service.

The simulator will be modified as part of the plant modifications so that it reflects the as-built condition in the plant for the systems that are controlled from the main control room.

The classroom and simulator training will be incorporated into the operator requalification training program. The frequency of the requalification training has not been established. The frequency of the operator training is determined by using the systematic approach to training (SAT) as described in the Employee Training and Qualification System (ETQS) Standards. The frequency of the training is based on a number of factors such as importance, difficulty, and how often the operators would be expected to operate the system in an emergency. Based on the results of the training analysis, the frequency of requalification could be every 2 years or every 4 years.

RAI 40 [T/H]

Provide the following information for the tornado and HELB restoration procedures:

1. How will the restoration procedures be laid out?
2. Will the procedures be symptom-based such that it would be able to be used for any tornado or HELB?
3. When will they be completed and implemented.

Duke Energy Response

The "restoration procedures" referred to in the RAI are those procedures that would be used to assess damage to plant equipment and the associated repair actions that may be necessary due to a tornado or a postulated HELB inside the Turbine Building that requires the use of the PSW or SSF system for the establishment and maintenance of safe shutdown.

Damage assessment and repair procedures were initially created for managing the damage caused by an Appendix R fire scenario where the SSF was relied upon for the establishment and maintenance of safe shutdown. However, these procedures would be utilized to determine the extent of damage resulting from a tornado or a postulated HELB in the Turbine Building and to direct repair efforts to restore needed functions. These procedures are coordinated in that an initial damage assessment is performed to determine the capability of restoring electrical power and plant systems needed to support a plant cooldown. Once the extent of damage has been identified, damage repair procedures are initiated to either restore that equipment or provide an alternate means of providing the system function.

These procedures are not symptom based. However, the damage assessment procedure is a systematic approach to determining the status of systems needed for plant cooldown. Therefore, the actions would be taken regardless of the event causing the damage.

The existing damage assessment procedure does not specifically check for piping integrity. Procedure enhancements are planned to improve the damage assessment to identify any breaches of HPI, LPI, CCW, and LPSW (essential headers) so that they can either be isolated or repaired prior to placing the systems in service. This has been entered into Oconee's corrective action program.

RAI 41 [T/H]

In the licensee's October 23, 2009, supplement the licensee refers to agreements with the NRC staff on a number of different issues concerning tornado/HELB mitigation strategies. The NRC staff notes that that these agreements have not been submitted to the NRC in accordance with 10 CFR 50.4. Please provide the documentation in accordance with 10 CFR 50.4.

Duke Energy Response

In the 2006-2007 timeframe, a number of pre-submittal meetings were conducted in support of planned tornado and HELB LAR submittals. Common understanding of the approaches to be used in the submittals was achieved for the issues anticipated. The common understandings achieved were compiled in a Tornado, HELB, or Common Item matrix and docketed (ADAMS acquisition number ML70670203).

The NRC items used in these matrices were a compilation of past NRC tornado and HELB issues taken from a number of past communications between Duke Energy and the Staff that were being discussed in pre-submittal meetings in order to develop a common understanding of the licensing approaches to be included in the tornado and HELB license amendment requests. The information discussed in these meetings was not intended to be used for regulatory decision making relative to Oconee's license. As a result, each matrix was tagged with "Draft Document - For Discussion Purposes Only" in the footer of each page and placed on the docket as attachments to the NRC meeting summaries.

Based on the age of the information and the nature of the previous correspondence, Duke Energy is not able to complete internal processes to validate the completeness and accuracy of the information presented in the matrices. Where a common understanding was achieved that resulted in an agreement to provide additional information in a license amendment request, the matrices have been updated to indicate the location of Duke Energy's position regarding the matter in the tornado and/or HELB license amendment requests that were submitted in accordance with 10 CFR 50.4. The Attachment to this submittal contains the updated matrices.

RAI 42 [T/H]

Describe how the PSW system will be isolated from other safety systems and/or will not cause unintended results should a PSW system active or passive failure occur. Include information for both the electrical and mechanical system associated with the PSW installation.

Duke Energy Response

The PSW System is a backup system utilized following postulated HELB or Tornado events, when safety systems, located in the Turbine Building, are not available to support a safe shutdown of the Oconee Units. The PSW system is divided into a mechanical portion and an electrical portion. A discussion of each of these parts of the PSW System follows:

Mechanical Portion

The mechanical portion of the PSW System consists of the PSW Booster Pump, the PSW Primary Pump, the associated valves, piping, instrumentation, the PSW Pump Room Exhaust Fan, and the PSW Building HVAC System. The mechanical portion of the PSW System is isolated both physically and functionally from other safety systems. The PSW Booster Pump, the PSW Primary Pump, the pump suction & discharge piping, and the associated vales & instrumentation for the pumps are located in the PSW Pump Room. This room is physically isolated from other equipment in the Unit 2 Auxiliary Building. The only mechanical PSW equipment located in the Auxiliary Building beyond the PSW Pump Room are the piping lines to each unit's Emergency Feedwater (EFW) System, the piping lines to the Low Pressure Service Water (LPSW) piping, and the PSW Pump room Exhaust Fan. The PSW System pump suction line interfaces with the Unit 2 Condenser Circulating Water (CCW) System. The PSW Building HVAC System is located within the PSW Building. The PSW System is functionally isolated from the EFW and LPSW Systems by at least one closed valve. The suction line for the PSW Pumps from the CCW System has a normally open isolation valve.

The PSW System components in the Auxiliary Building are maintained in a standby condition with no equipment operating. Since the PSW System is not in operation and no mechanical movement of any component is required to maintain the standby condition, no single active failures (See "Single Active Failure" definition Page 1-16 of the ONS HELB Report, ONDS-351) can be created. This condition of no mechanical movement also applies to the electrical power and control circuits for the pumps and valves, and thus, creates no single active failures. If the PSW System is manually actuated, then the EFW and/or the LPSW Systems are not available to support the safe shutdown of the Oconee units, and any single active failure of the PSW System could not adversely affect either of these systems. Also, because of the physical and functional isolation and the standby mode status of the PSW System, there are no passive failures of the PSW System that could adversely affect either of these systems. In order to provide an assured water source the PSW Pumps suction line is required to be open to support operation of the PSW Pumps. The design of the PSW Pumps suction piping and connection to the CCW pipe has been made based upon the need to have the PSW Pumps suction line non-isolated from the CCW System.

The PSW Building HVAC System is in continuous operation. However, the PSW Building is physically remote from the other areas of the ONS. As such, any single active failure or passive failure of the PSW Building HVAC System could not adversely affect any other safety system of any Oconee Unit.

Electrical Portion

The major on-site electrical components of the PSW system are located in the PSW building, the Auxiliary Building, the Keowee Hydro Station and the Standby Shutdown Facility (SSF) and are summarized as follows:

In the PSW building, the major PSW electrical components consists of medium voltage switchgear which is composed of two 10 MVA 13.8/4.16 kV transformers and 13.8 and 4.16 kV breakers, a 600 VAC load center with a 5 MVA 4160/600 VAC transformer, a 600 VAC motor control center, a 600 VAC manual transfer switch, a SSF submersible pump starter, two 125 VDC station batteries, two battery chargers, one 125 VDC distribution center, 125 VDC panelboards and associated miscellaneous electrical support equipment including a 600/208/120 VAC transformer, 208/120 VAC panelboards, lighting, receptacles, a HVAC system and a fire detection system.

In the Auxiliary Building, the major PSW electrical components consists of 600 VAC motor control centers, 600 VAC pressurizer heater manual transfer switches, 4.16 kV manual alignment switches and motor operated HPI pump motor transfer switches, 600 VAC automatic transfer switches for the Vital I&C normal battery chargers, 600/208/120 VAC transformers, 208/120 VAC panelboards, 125 VDC panelboards and associated local and main control room instrumentation and controls

At the Keowee Hydro Station, the major PSW electrical components consist of 13.8 kV switchgear, junction boxes for connecting to the existing 13.8 kV buss and associated Keowee and main control room instrumentation and controls.

At the SSF, the major PSW electrical components consist of a 4.16 kV switchgear cubicle and associated SSF control room instrumentation and controls.

The main electrical connections to existing safety systems are the HPI pump motor power feeds, the Vital I&C normal battery chargers, the pressurizer heaters (a non-QA-1 electrical system), Keowee 13. 8 kV power and SSF 4.16 kV power.

Electrical isolation from existing safety systems is provided by transfer switches, breakers and/or fuses. A Failure Modes and Effects Analysis/Single Failure Analysis (FMEA/SFA) will be performed for the PSW electrical system. The FMEA/SFA will evaluate the PSW electrical system as a whole and the interfaces with existing systems including HPI pump motor power, normal Vital I&C battery charger input power, pressurizer heater power and the Keowee and SSF power systems.

The FMEA/SFA is performed as part of the Engineering Change process in accordance with EDM-105 (Guidelines for Performing a Failure Mode and Effects Analysis and Single Failure Analysis). The FMEA evaluates the PSW design requirements related to redundancy, failure detection systems, fail-safe characteristics, and automatic and manual override. The SFA (if required) is performed similar to the FMEA but that the failure modes are a function of licensing basis requirements.

RAI 43 [T/H]

Provide the following information for the new protected service water (PSW) system transformer, switchgear, load center and the circuit breakers: (1) equipment design ratings, (2) a summary of the analyses performed to show the loading, short circuit values and the interrupting ratings, voltage drop, and protection and coordination, (3) the existing station ASW switchgear ratings, and (4) the periodic inspection and testing requirements for electrical equipment. Provide applicable schematic and single line diagrams.

Duke Energy Response

Please refer to response for RAI 2-27 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 44 [T/H]

Provide the following information concerning the proposed PSW instrumentation and control (I&C) power and the interface with the existing plant vital I&C power: (1) design of the direct current (DC) system for the PSW system including how the DC control power for the new PSW load center, switchgear and the transformer will be provided, (2) the impact on existing DC vital system including loading on the existing battery and the battery charger, (3) describe the analysis performed to determine the capacity of the batteries and the battery charger, voltage requirements at the equipment terminals, electrical protection and co-ordination, and (4) the periodic inspection and testing requirements. Provide applicable schematic and single line diagrams.

Duke Energy Response

Please refer to response for RAI 2-28 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 45 [T/H]

The Keowee Hydroelectric Units (KHUs) will provide power supply to the PSW switchgear through underground cables. Provide analyses to show the kilo volt ampere (kVA) loading, new circuit breaker rating, short circuit values, and voltage drop. In addition, provide information on the electrical protection and coordination, and the periodic inspection and testing requirements. Further, explain how the redundancy and independence of the Class 1E power system is maintained as a result of the proposed modification. Provide applicable schematic and single line diagrams.

Duke Energy Response

Please refer to response for RAI 2-29 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 46 [T/H]

The PSW system will be fully operational from the respective unit's main control room and will be activated when existing redundant emergency systems are not available. Describe how the alarms, indications, and the electrical controls will be provided from the main control rooms of Units 1 and 2 to the proposed PSW switchgear. Explain how the controls are provided for Unit 3. Provide applicable electrical schematics and evaluations highlighting the design features.

Duke Energy Response

Please refer to response for RAI 2-30 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 47 [T/H]

Provide information on how the licensing basis for physical independence and separation criteria are met for the PSW electrical system.

Duke Energy Response

Please refer to response for RAI 2-31 contained in Duke Energy RAI submittal dated June 24, 2010 (Ref. 6).

RAI 48 [T/H]

The new PSW system switchgear will receive power from the KHUs via a tornado-protected underground feeder path. Provide the following information:

1. Type of underground cable installation, i.e., direct burial or in duct banks, manholes etc.
2. How the licensee will ensure that the proposed new underground cables remain in an environment that they are qualified for
3. Periodic inspections and testing planned for cables to monitor their performance, and
4. Details regarding cable size, type, maximum loading requirements, and cable protection devices.

Duke Energy Response

Please refer to response for RAI 2-32 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 49 [T/H]

Provide information concerning the design details for the new 100/13.8 kV substation, the PSW transformer and switchgear building power feeds, its protection, controls and alarms features. Provide applicable single line diagram and electrical schematics.

Duke Energy Response

Please refer to response for RAI 2-33 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 50 [T/H]

Two new power feeds will be installed to the auxiliary building (AB) with one power supply to the Units 1, 2, and 3 AB equipment high-pressure injection (HPI) pumps and vital I&C normal battery chargers and other power supply to the backup power to the Units 1, 2, and 3 pressurizer heaters. Provide the following information concerning this installation: (1) compare and contrast the existing power supply requirements for the above loads, (2) how the electrical separation, independence, and redundancy requirements are maintained, (3) summary of the voltage analyses for the equipment/components affected by this modification, (4) design details for the new power feeds to AB, (5) periodic inspections and testing schedule for the these cables to monitor their performance, and (6) provide the electrical schematics and one-line drawings for these power feeds.

Duke Energy Response

Please refer to response for RAI 2-34 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 51 [T/H]

Provide confirmation that the maximum float/equalizing voltage does not exceed the equipment maximum dc voltage rating.

Duke Energy Response

Please refer to response for RAI 2-35 contained in Duke Energy RAI submittal dated June 24, 2010 (Ref. 6).

RAI 52 [T/H]

Describe in detail how the 125 vdc vital I&C primary and backup power cables and the KHU emergency start circuitry will be rerouted from the turbine building to the auxiliary building.

Duke Energy Response

Please refer to response for RAI 2-36 contained in Duke Energy RAI submittal dated August 31, 2010 (Ref. 5).

RAI 53 [T/H]

To ensure licensing-basis clarity and component operability, Technical Specifications (TSs) need to properly address the tornado mitigation systems (e.g., PSW/SSF, protected service water/ standby shutdown facility, etc.) in a manner that is consistent with the Standard TS requirements that have been established for the functions that are being performed by these systems. For example, the minimum required mission time should be 7 days and the Completion Times should be limited to 72 hours in most cases for the SSF and the PSW including maintenance. Justify the existing limiting condition for operation (LCO) time for the SSF in the current TSs and the proposed LCO for the PSW system based on the fact the proposed tornado mitigation strategy relies solely on the SSF and the repair of the PSW system to achieve and maintain hot standby and entry into cold shutdown following a design basis tornado/HELB. The proposed TS change for the PSW system and the existing SSF does not preclude both diverse systems being out of service concurrently please provide a justification for this.

Duke Energy Response

For tornado response please refer to RAI response letter dated August 31, 2010 (Ref. 5).

HELB Background

As ONS construction was nearing completion, the Atomic Energy Commission (AEC) issued a letter from A. Giambusso (AEC), Deputy Director for Reactor Projects Directorate of Licensing, to Duke Power Company (now Duke Energy Carolinas, LLC (Duke Energy), dated December 15, 1972[2]. The "Giambusso Letter" required licensees to address the consequences of pipe ruptures outside containment and submit their analyses to the AEC for review. Due to the specific guidance in the letter, the applicable events were identified as "High Energy Line Break" (HELB) events. The "Giambusso Letter" was amended by an errata sheet provided in a letter from A. Schwencer (AEC), Chief Pressurized Water Reactors Branch No. 4 Directorate of Licensing, to Duke Power Company, dated January 17, 1973[3] (the "Schwencer letter").

Duke Energy's evaluations of postulated pipe ruptures outside containment were documented in MDS Report No. OS-73.2 dated April 25, 1973, with Supplement 1 to the report dated June 22, 1973 and Supplement 2 to the report dated March 12, 1974. The final report is referred to herein as "current HELB report," "MDS Report" and/or "OS-73.2."

The MDS report was incorporated into the ONS license application by reference. It was subsequently approved and accepted by the AEC. "Safety Evaluation prepared by the Directorate of Licensing related to the Oconee Nuclear Station, Units 2 and 3," (referred to herein as "the SER") dated July 6, 1973[4], was issued as part of the initial licensing of Units 2 and 3. SER Section 7.1.11 "High-energy Line Rupture External to the Reactor Building" addressed the MDS report, and Attachment E of the SER repeated the NRC HELB criteria, as amended by the Schwencer letter. The following is extracted from Section 7.1.11:

"The basic criteria require that:

(1) Protection be provided for equipment necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, from all effects resulting from ruptures in pipes carrying high-energy fluid, up to and including a double-ended rupture of such pipes, where the temperature and pressure conditions of

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[2] Letter dated 15 December 1972 from A. Giambusso (AEC) to A. C. Thies (DPC) transmitting the "General Information Required for Consideration of the Effects of a Piping system Break Outside Containment."

[3] Clarification Letter (related to the 15 December 1972 letter), dated 17 January 1973, from A. Schwencer (AEC) to A. C. Thies (DPC)

[4] Safety Evaluation Report (From AEC) for Oconee Units 2 & 3, July 6, 1973.

the fluid exceed 200 °F and 275 psig. Breaks should be assumed to occur in those locations specified in the "pipe whip criteria." The rupture effects on equipment to be considered include pipe whip, structural (including the effects of jet impingement) and environmental.

(2) Protection be provided for equipment necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protected equipment, from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying high-energy fluid routed in the vicinity of this equipment, where the temperature and pressure conditions of the fluid exceed 200 °F and 275 psig. The size of the cracks should be assumed to be l/2 the pipe diameter in length and l/2 the wall thickness in width....

Staff Evaluation and Conclusion

The staff has evaluated the assessment performed by the applicant and has concluded that the applicant has analyzed the facilities in a manner consistent with the intent of the criteria and guidelines provided by the staff. The staff agrees with the applicant's selection of pipe failure locations and concludes that all required accident situations have been addressed appropriately by the applicant.

Furthermore the staff has evaluated the analytical methods and assumptions used in the applicant's analyses and find them acceptable and concurs with the proposed plant modifications and the criteria to be used in their designs."

Several years after approval of the MDS report and initial licensing of ONS, the SSF was constructed. The SSF provides additional defense-in-depth protection to achieve and maintain Mode 3 with an average Reactor Coolant System temperature ≥ 525 °F following postulated fire, sabotage, or internal flooding events.

The SSF Reactor Coolant Make-up (RCMU) system is the SSF sub-system designed and credited to supply Reactor Coolant Pump seal injection flow in the event that the High Pressure Injection (HPI), the normal make up system, becomes inoperable when a Unit's RCS temperature is > 250 °F. It can recover RCS volume shrinkage caused by cooling the RCS to Mode 3 with an average Reactor Coolant temperature ≥ 525 °F. However, the SSF Reactor Coolant Make-up System is not credited for events, such as LOCA, which result in significant loss of RCS inventory. The SSF Auxiliary Service Water System (ASW) is the SSF sub-system credited as the backup to the Feedwater (FDW) and Emergency Feedwater (EFW) systems.

A 1998 Duke HELB self-assessment revealed issues with the original OS-73.2 report, and as a result, Duke decided to fully revalidate and revise the HELB CLB. In late 1999, Duke initiated a project to determine scope of these CLB revision efforts[5]. This HELB CLB revision effort was completed on a unit by unit basis with the final license amendment request submitted to the Staff in June 2009.

The updated HELB mitigation strategy addresses the measures to be taken to minimize postulated pipe failures that could affect structures, systems, and components (SSC) necessary to achieve and maintain a Safe Shutdown condition. SSCs located in the TB are protected from postulated breaks and cracks that could occur in the AB due to separation. In addition, SSCs located in the AB are protected from postulated breaks and cracks that could occur in the TB. SSC's located in the Standby Shutdown Facility (SSF) are protected from breaks that could occur in the TB. SSF related systems and components located in the WPR have been evaluated for pipe ruptures postulated to occur in the WPR.

[5] Letter from W. R. McCollum, Jr., Vice President, Oconee Nuclear Station, to the Nuclear Regulatory Commission, "High-Energy Line Break outside Reactor Building Methodology," dated July 3, 2002.

Technical Specification discussion

The SSF and the PSW systems do not mitigate design basis accidents (DBAs). Consequently, neither SSF nor PSW operability requirements readily fit into the standardization process established by NUREG 1430 [Standard Technical Specifications for Babcock and Wilcox Plants], i.e., the document does not contain any criteria for a protected service water system. The system in the NUREG with the closest fit is EFW, however, EFW requirements are tied to the mitigation of DBAs.

The existing limiting conditions for operation (LCOs) for the SSF systems and the proposed LCOs for the PSW systems are justified based on the overall risk improvement that is achieved by the installation of the PSW system. The current ONS tornado licensing basis relies on redundant and diverse secondary side decay heat removal (SSDHR) sources and primarily on the tornado-protected SSF Auxiliary Service Water system for acceptability. In order to simplify the licensing basis, the requested tornado licensing amendment credits only the SSF for mitigation of tornado-related damage to the station. The redundancy and diversity of SSDHR sources remains unchanged and is in fact improved by the addition of the PSW system. The availability of the redundant, diverse SSDHR sources (including PSW) are (or will be) in the station's PRA. As a result, the SSF system's risk worth is expected to decrease with the completion of the PSW project. A decrease in risk worth of the SSF systems justifies the continued acceptability of the exiting SSF LCOs.

RAI 54 [T/H]

Provide a list of any new analyses, codes, and/or models being utilized for the proposed tornado/HELB mitigating strategies that need to be integrated into the UFSAR. Provide the justification for their use.

Duke Energy Response

For tornado response please refer to RAI response letter dated August 31, 2010 (Ref. 5).

The new analyses, codes and models utilized in evaluating the consequences of postulated HELBs outside containment are discussed in the HELB Report. The HELB Report will be included by reference to section 3.6.1 of the UFSAR. The new analyses, codes and models will not be included in Chapter 15 of the UFSAR.

A number of computer codes were utilized to evaluate the effects of HELBs. These include:

- RETRAN-3D was used to determine the plant response following a postulated letdown line break and a double main steam line break.
- RETRAN-02 was used to determine the plant response following a postulated main feedwater line break
- SIMULATE-3P was used to determine if a return-to-power condition resulted following a postulated double main steam line break
- CASMO-3 was used in conjunction with SIMULATE-3P.
- VIPRE-01 was used to determine the minimum DNBR following a postulated double main steam line break
- LOCADOSE was used to determine potential offsite and control room doses following a postulated letdown line break.
- Gothic 4.0 was used to calculate pressure and temperature response inside the penetration rooms following a main steam line break or main feedwater line break.

- Gothic 7.0 was used to calculate the pressure response in other enclosed areas of the auxiliary building with postulated breaks in the plant heating system.
- RT^3 was used to evaluate the temperature response of the control complex following a main steam line break in the penetration room.
- A model was developed for the Turbine Building to evaluate the effects of pipe whip on the structure using GT (Georgia Tech) STRUDL.

The use of computer codes RETRAN-3D, RETRAN-02, SIMULATE-3P, CASMO-3 and VIPRE-01 are described in Section 15.1.2 of the UFSAR.

LOCADOSE computer code has been previously used to determine offsite and control room doses for design basis events in Chapter 15 of the UFSAR.

The GOTHIC 4.0/DUKE computer code was first licensed for use in the Duke methodology report DPC-NE-3004-PA, "Mass and Energy Release and Containment Response Methodology" for Duke's McGuire / Catawba units in 1995. In this report, the use of the GOTHIC 4.0/DUKE code and models for pressure/temperature transient calculations within the ice condenser containments of McGuire and Catawba was justified. It has been used for many Duke licensing basis applications as described in Section 6.2 of the UFSARs for those plants. One recent application using this computer code is the License Amendment Request (LAR) to implement the Water Management initiative at Catawba and McGuire. An SER was received from the NRC for Catawba's Water Management LAR on June 28, 2010. The corresponding LAR for McGuire is still under review.

Other LARs in which the GOTHIC 4.0/DUKE code was applied have been submitted and approved since 1995. These include the changes to Tech Spec Surveillance Requirement (SR) 3.6.12.4 to allow asymmetric loading of the ice condenser baskets (lower allowable weights for inner rows), the elimination of the torque test on the ice condenser lower inlet doors from SR 3.6.13.6, and changes to allow the manual starting of the Containment Air Return Fans for certain Small Break LOCA scenarios. Again, these LARs were submitted for the McGuire/Catawba units.

GOTHIC 4.0/DUKE was the latest version of GOTHIC released by Numerical Applications Inc. (the code vendor) at the time the DPC-NE-3004 submittal was made. Several updated code releases have been released since then, with the most recent GOTHIC release being Version 7.2e. In September 2003, an SER was received for Revision 1 of the Duke Methodology Report DPC-NE-3003. In this revision of Oconee's containment analysis methodology report, the GOTHIC 7.0 code was approved for use "to calculate the reactor building response to high energy line breaks". The GOTHIC 7.0 code, which is a later version of GOTHIC 4.0/DUKE, was demonstrated to provide similar results to the FATHOMS/DUKE-RS code which had been Oconee's licensing basis containment analysis methodology since the original approval of DPC-NE-3003 in 1994.

The high energy line break response inside the penetration rooms described in Duke calculation OSC-8104 is a very similar calculation to the LOCA/SLB responses inside the Oconee Reactor Building addressed in DPC-NE-3003. The GOTHIC 4.0/DUKE code is expected to give similar answers to any similar analysis performed by GOTHIC 7.0, as there have been no major changes made to the GOTHIC code between those releases which would significantly impact the results of such an analysis.

Therefore, it is demonstrated that the GOTHIC 4.0/DUKE code is an acceptable method for calculating conditions within the Oconee penetration rooms following a high energy line break.

The RT^3 computer code is currently used to evaluate the temperature response of the control complex following a loss of ventilation. The same model was employed with a new heat load source from the penetration room due to the MSLB.

RAI 55 [T/H]

Provide the following information concerning the ability to achieve and maintain hot standby (TS MODE 3) following the worst-case design basis tornado/HELB.

a. List of equipment that will be used
b. Initial plant conditions.
c. Discuss any scenarios where with use of only the SSF/PSW to achieve and maintain hot standby would cause any of the units to operate outside the normal operating boundaries as described in the UFSAR (i.e., the RCS does not stay sub-cooled with a pressurizer steam bubble).
d. Provide the basis for the SSF/PSW initiation times and confirmation that human factors assessment has been completed that is consistent with the NRC review standards and guidance to validate operator actions and times.
e. Provide a list of all operator actions, a timeline for achieving hot standby and the systems that will be available and the amount time when other systems (SSF/PSW/HPI) will have to be repaired/restored to maintain the units in a safe and stable condition following a tornado/HELB.

Duke Energy Response

For tornado response please refer to RAI response letter dated August 31, 2010 (Ref. 5).

The "worst-case" HELBs are described in Section 7.0 of ONDS-351. Three classes of HELBs are described in this section. They include Main Steam Line Breaks (MSLB), Main Feedwater Line Breaks (FWLB) and a postulated line break in the Reactor Coolant letdown line. These events were analyzed since they would generate the most extreme transient conditions in the station. The list of equipment used to mitigate the consequences of these events is described for each event. The initial plant conditions are also described for each event.

There are HELB scenarios (i.e., MSLBs) where the RCS sub-cooling margin may be lost. These are also discussed in Section 7.0 of ONDS-351. These scenarios did not credit the SSF for event mitigation.

The bounding scenario for the SSF/PSW initiation times is a postulated FWLB inside the turbine building that results in a loss of main and emergency feedwater with a loss of all reactor coolant pump seal cooling. A source of feedwater to the steam generators must be re-established within 15 minutes. A source of reactor coolant pump seal cooling must be re-established within 20 minutes to avoid damaging the seals. These actions will ensure that RCS natural circulation flow is not interrupted due to excessive voiding in the RCS hot legs.

The SSF ASW system is capable of being aligned within 14 minutes. The initiation time for SSF ASW has been previously reviewed and approved by the NRC. The PSW system will be capable of being aligned within 15 minutes, once installed. Operator time validations are completed to confirm the capability as part of the modification process.

The SSF RC Make-Up System is capable of being aligned to provide seal injection flow within 20 minutes. The initiation time for the SSF RC Make-Up System has been previously reviewed and approved by the NRC. Seal injection flow from an HPI pump (powered from PSW) will be capable of being established within 20 minutes, once installed. Operator time validations are completed to confirm the capability as part of the modification process.

If the SSF is the only means available for establishing safe shutdown following a postulated HELB or tornado, then some equipment would need to be restored within 72 hours to maintain safe shutdown. The selection of equipment to restore would depend on the extent of damage resulting from the postulated HELB or tornado. Normal plant systems located inside the Turbine Building would be the preferred choice. However, if the extent of plant damage would preclude their recovery within 72 hours, the PSW mechanical and electrical systems would need to be restored.

If the PSW system is being utilized to maintain safe shutdown following a postulated HELB, there is no immediate need for plant cooldown. However, the PSW system is capable of supporting a plant cooldown to approximately 250°F. Operators would need to be dispatched to throttle the atmospheric dump valves to enable a plant cooldown. RCS inventory control and RCS pressure control could be maintained inside the main control room.

RAI 56 [T/H]

Discuss how cold shutdown will be achieved following a design basis tornado/HELB including, (a) define the actions necessary for achieving cold shutdown based on the worst-case repairs that will need to be made following a tornado/HELB; (b) recognition of the strategy/systems to be used (e.g., residual heat removal, PSW, SSF, LPI, HPI, pressurizer heaters, atmospheric dump valves, instruments, etc.) identification of specific vulnerabilities that need to be addressed, equipment to be staged (e.g., cable, motors, motor control centers, switchgear portable pumps etc.); and, (c) a human factors assessment of effort/repair that is consistent with the NRC review standards/guidance. Provide a time line and procedures for maintaining safe and stable conditions after entering into hot standby and an estimate for achieving cold shutdown. Provide the limitations of the SSF/PSW systems if these are the only systems used to achieve hot standby and maintain safe and stable conditions.

Duke Energy Response

For tornado response please refer to RAI response letter dated August 31, 2010 (Ref. 5).

A discussion is provided in ONDS-351 regarding achieving cold shutdown following postulated HELBs outside containment. There are three general areas where HELBs are postulated to occur. The first general area is inside the Auxiliary Building. The second general area is outside in the yard. The third general area is inside the Turbine Building. Postulated HELBs inside the Auxiliary Building or outside in the yard would not prevent the establishment of cold shutdown. However, some postulated HELBs inside the Turbine Building could result in damage to plant equipment where cold shutdown could not be established without damage repairs.

There are some postulated HELBs inside the Turbine Building where all 4160VAC power could be lost to each unit. For these postulated HELBs, either the PSW System or SSF are utilized to establish and maintain safe shutdown. The PSW system in conjunction with the atmospheric dump valves is capable of supporting a plant cooldown to approximately 250°F without damage repairs. However, power must be restored to the core flood tank outlet valves when RCS pressure is reduced to less than 800 psig to enable their closure. Damage repairs are necessary to restore CCW, LPSW and LPI to operation to enable RCS cooldown from approximately 250°F to cold shutdown. The CCW system is restored by repowering one CCW pump motor from an available 4160VAC power source, providing cooling water to the CCW pump motor, and opening two condenser waterbox outlet valves using bottled nitrogen if instrument air is not available. The LPSW system is restored by repowering one LPSW pump motor shared by Units 1 and 2, and one LPSW pump motor for Unit 3 from an available 4160VAC power source. The LPI system is restored by restoring 600VAC power to LP-1 and LP-2 (which are located inside the Reactor Building) in addition to repowering one LPI pump per unit from an available 4160VAC power source.

There are some postulated HELBs inside the Turbine Building where interactions with Condenser Circulating Water (CCW) and Low Pressure Service Water (LPSW) piping may result in flooding of the Turbine Building resulting in a total loss of emergency feedwater and low pressure service water. Additional damage repairs would be necessary to restore CCW and LPSW to operation. These repairs would include isolation of broken piping where possible, piping repair for breaks that cannot be isolated, and replacement of the LPSW pump motors due to flooding.

The equipment needed to repower the 4160VAC CCW pump and LPSW pump motors is stored onsite. Spare motors for the LPSW pumps are stored onsite. The equipment needed to repower the 600VAC core flood tank discharge valves and the LPI valves located inside containment is also stored onsite. The equipment needed for damage repair is located external to the Turbine Building and will be available following the postulated HELBs inside the Turbine Building.

Regarding human factor assessments for the damage assessments and repairs described above, the actions taken to restore vulnerable equipment needed for a unit cooldown to approximately 250°F would be accomplished utilizing station procedures. The preparation, review and approval of station procedures are performed in accordance with section 17.3.2.14 of the Duke QA Topical Report. The damage assessment and repair procedures are classified as permanent technical procedures that would be utilized after the plant has been brought to a safe shutdown condition using the existing emergency procedures. The purpose of the damage assessment procedures is to determine the availability of unprotected systems and components utilized during a plant cooldown. The repair procedures are employed to restore any damaged equipment discovered during the assessment. The assessment and repair procedures would be initiated after the Emergency Response Organization (ERO) has been staffed. A validation and verification process is in-place to address the adequacy of technical procedures. These procedures are designated as the following types:

- Emergency Response Procedures (RP)
- Operating Procedures (OP)
- Instrument and Electrical Procedures (IP)
- Mechanical Maintenance Procedures (MP)

The time line for maintaining safe shutdown conditions using the PSW or SSF system is described in section 3.1 of ONDS-351. The description and capabilities of the PSW and SSF systems are described in sections 3.2 and 3.3 of ONDS-351, respectively.

RAI 57 [T/H]

Describe what instrumentation will be available following the worst case tornado/HELB. Describe all instrument failures (e.g., pressurizer level, etc.) and how they will be discerned in support of main control room and/or SSF/PSW control.

Duke Energy Response

For tornado response please refer to RAI response letter dated August 31, 2010 (Ref. 5).

Section 3.10 of ONDS-351 provides a list of indications needed by the operator to establish and maintain safe shutdown. The list of instrumentation included instrumentation needed in the main control room and the SSF control room. The list was used as potential safe shutdown "targets" in the evaluation of postulated HELBs outside containment. The instrumentation remains available following the worst case HELBs outside containment. Power for the main control room instrumentation following the worst case HELB is provided by the applicable unit's control batteries via the PSW electrical system, while the SSF control room instrumentation is provided by the SSF electrical distribution system.

The available indications inside the main control room for maintaining safe shutdown include:

- RCS Pressure
- RCS Temperature (hot legs, cold legs, and core exit thermocouples)
- Pressurizer Water Level
- RCS Water Levels (Reactor Vessel and Hot Legs)
- RCS Subcooling Margin
- Neutron Flux
- Letdown Storage Tank Water Level
- Borated Water Storage Tank Water Level
- High Pressure Injection Flow
- RCP Seal Injection Total Flow
- Steam Generator Water Level
- Steam Generator Pressure
- Upper Surge Tank Level (used only with emergency feedwater operation)
- Emergency Feedwater Flow
- Protected Service Water Flow

The available indications inside the SSF control room for maintaining safe shutdown include:

- RCS Pressure
- RCS Temperature (hot legs and cold legs)
- Pressurizer Water Level
- RC Makeup Pump Pressures (suction and discharge)
- RC Makeup Pump Flow
- Steam Generator Water Levels
- SSF Auxiliary Service Water Flow (to each unit).

RAI 58 [T/H]

Discuss the how the RCP seals are protected following a tornado/HELB.

Duke Energy Response

For tornado response please refer to RAI response letter dated August 31, 2010 (Ref. 5).

RCP seal cooling is normally provided by the High Pressure Injection (HPI) and the Component Cooling (CC) systems. Either system is capable of providing adequate RCP seal cooling to protect the seals. The CC system is not a high energy system. Therefore, no HELBs are postulated in the CC system. The HPI system is a high energy system. The CC system is credited for providing RCP seal cooling following a postulated HELB in any of the HPI seal injection lines. There are no postulated HELBs inside the Auxiliary Building that would disable both the HPI and CC systems. However, there are postulated HELBs inside the Turbine Building that may result in the loss of both HPI and CC systems. For the postulated HELBs that result in the loss of both HPI and CC systems, the PSW system is credited for providing power to one HPI pump, HPI valves and instruments needed to support RCP seal injection. These power alignments can be made from the control room with RCP seal injection reestablished within 20 minutes. The PSW system and the associated power sources are protected from the effects of postulated HELBs inside the Turbine Building.

In addition to the PSW system, the SSF Reactor Coolant Makeup (RCMU) system is capable of providing RCP seal cooling by means of seal injection within 20 minutes after a loss of both HPI and CC systems due to postulated HELBs. The SSF RCMU system is protected from the effects of postulated HELBs inside the Turbine Building.

Since seal injection flow is established within 20 minutes after a loss of HPI seal injection and CC system flow, seal degradation or failure will not occur and flow rates associated with a seal loss of cooling accident (LOCA) will not occur.

The SSF RCMU system is the credited system for providing RCP seal cooling following tornado damage that result in the loss of both HPI and CC systems.

RAI 59 [T/H]

It has been noted by the NRC staff that the LARs for both tornado and HELB mitigation strategies contain information that appears not up to date (i.e. commitments, commitment dates, system designs, complete documentation, HELB report) Please review the LARs and all supplemental submittals and update the LARs as necessary.

Duke Energy Response

In an effort to expedite the HELB LAR submittal process it was decided that each unit would be evaluated and submitted separately until the HELB mitigation strategy report was completed for the entire station, i.e., the HELB report submitted as part of the third LAR installment not only included that unit but also enveloped the results of the other two units.

The Unit 1 HELB LAR was submitted in June 2008, followed by Unit 2 HELB LAR in December 2008, and finally the Unit 3 HELB LAR was submitted in June 2009.

The complete HELB licensing footprint is comprised of the following:

1. Unit 3 LAR containing the complete HELB Report (Rev. 2),
2. Unit 3 LAR technical specifications,
3. Unit 3 LAR UFSAR update,
4. Unit 3 LAR commitment table,
5. Unit 3 LAR tables and figures,
6. Unit 1 LAR commitment table
7. Unit 1 LAR tables and figures,
8. Unit 2 LAR commitment table,
9. Unit 2 LAR tables and figures,
10. Duke Energy submittal of supplemental information dated September 2, 2009,
11. Duke Energy RAI response dated October 23, 2009.

The complete Tornado licensing footprint is comprised of the following:

1. Tornado LAR dated June 26, 2008,
2. Tornado LAR UFSAR update,
3. Tornado LAR commitment table,
4. Duke Energy RAI response dated September 2, 2009,
5. Duke Energy RAI response dated May 6, 2010.
6. Duke Energy RAI response dated June 24, 2010,
7. Duke Energy RAI response dated August 31, 2010.

The following are the commitments associated with either the HELB (H) or Tornado (T) LARs.

Note: The sequential numbering of these commitments includes, but does not show, prior commitments made in Duke Energy's November 30, 2006 letter and subsequent update letters. From the lists below, commitments 18T and 19T were revised to coincide with the anticipated approval of the tornado mitigation strategy LAR. In addition, for both the tornado and HELB mitigation strategies, licensing actions associated with the incorporation of Fiber Reinforced Polymer and Main Steam Isolation Valves (MSIVs) for applicable events are or will be addressed by separate applications.

No.	UNIT 1 HIGH ENERGY LINE BREAK LAR COMMITMENTS – 26H THROUGH 33H	DUE DATE
26H	The inlet isolation valves to the Letdown Coolers on the Letdown Line (1HP-1 & 1HP-2) will be upgraded to permit their use following a postulated HELB on the Letdown Line at Containment Penetration No. 6. With these valves upgraded, either could then be closed if either of the inboard containment isolation valves (1HP-3 & 1HP-4) fails to close in order to mitigate the postulated HELB on the Letdown line.	To be provided to the Staff upon issuance of the SER.
27H	The ducting near the Control Complex is being upgraded with duct registers or cover plates to prevent the potential propagation of the HELB generated environment in the East Penetration Room to the Control Complex.	To be provided to the Staff upon issuance of the SER.
28H	The valves (1HP-103 & 1HP-107) on the individual suction lines to the "A" & "B" High Pressure Injection (HPI) pumps are being upgraded to allow the remote operation (operated outside the HPI pump room) of these valves. The remote operation of these valves allow the isolation of postulated HELBs on the discharge side of the HPI Pumps without compromising the availability of the other HPI Pumps and the need for maintaining the Letdown Storage Tank aligned to the HPI Pump suction piping. For a single active failure of either valve 1HP-103 or 1HP-107 to close, a redundant, remotely operated valve is provided on each of the HPI Pumps "A" and "B" to assure HELB mitigation.	To be provided to the Staff upon issuance of the SER.
29H	The position of several Plant Heating System isolation valves is being changed from "OPEN" to "CLOSED." This position change will eliminate the need to postulate Plant Heating System HELBs in the East Penetration Room and West Penetration Room, because these piping lines will be isolated during normal plant conditions of the station.	To be provided to the Staff upon issuance of the SER.
30H	Turbine Building structural support column D-26 will be modified by adding a brace to the column. This brace is necessary to prevent potential failure of the column, when subjected to a pipe whip load. This upgrade prevents the loss of the routing to get temporary cabling to the Low Pressure Injection and Low Pressure Service Water pump motors.	To be provided to the Staff upon issuance of the SER.
31H	The existing Condenser Circulating Water (CCW) discharge stop gates will be replaced and four (4) new stop gates will be obtained. These stop gates will be used to terminate all reverse flow through HELB damaged Low Pressure Service Water and CCW piping. This modification is required, in order to recover from a Turbine Building flood event caused by a postulated HELB therein.	To be provided to the Staff upon issuance of the SER.
32H	Evaluate the ability of the Standby Shutdown Facility to perform its safety functions with a compromised main steam pressure boundary due to potential breaks in the main steam system and other HELBs.	Complete.
33H	Weep holes will be installed in the bottom of the outside-containment junction box enclosures for the Viking Electrical Penetrations. Also, the electrical penetration inspection procedure is being amended to inspect the weep holes for blockage	Complete.

No.	UNIT 2 HIGH ENERGY LINE BREAK LAR COMMITMENTS – 34H THROUGH 39H	DUE DATE
34H	The inlet isolation valves to the Letdown Coolers on the Letdown Line (2HP-1& 2HP-2) will be upgraded to permit their use following a postulated HELB on the Letdown Line at Containment Penetration No. 6. With these valves upgraded, either could then be closed if either of the inboard containment isolation valves (2HP-3 & 2HP-4) fails to close in order to mitigate the postulated HELB on the Letdown line.	To be provided to the Staff upon issuance of the SER.
35H	The Unit 2 HVAC ducting near the Control Complex is being upgraded with duct registers or cover plates to prevent the potential propagation of the HELB generated environment in the East Penetration Room to the Control Complex.	To be provided to the Staff upon issuance of the SER.
36H	The valves (2HP-103 & 2HP-107) on the individual suction lines to the "A" & "B" High Pressure Injection (HPI) pumps are being upgraded to allow the remote operation (operated outside the HPI pump room) of these valves. The remote operation of these valves allow the isolation of postulated HELBs on the discharge side of the HPI pumps without compromising the availability of the other HPI Pumps and the need for maintain the Letdown Storage Tank aligned to the HPI Pump suction piping. For a single active failure of either valves 2HP-103 or 2HP-107 to close, a redundant, remotely operated valves is provided on each of the HPI Pumps "A" and "B" to assure HELB mitigation.	To be provided to the Staff upon issuance of the SER.
37H	The position of several Unit 2 Plant Heating System isolation valves is being changed from "OPEN" to "CLOSED." This position change will eliminate the need to postulate Plant Heating System HELBs in the East Penetration Room and West Penetration Room, because these piping lines will be isolated during normal plant conditions of the station.	To be provided to the Staff upon issuance of the SER.
38H	Turbine Building structural support Column D-29 & D-31 will be modified by adding a brace to the column. This brace is necessary to prevent potential failure of the column, when subjected to a pipe whip load.	To be provided to the Staff upon issuance of the SER.
39H	Weep holes will be installed in the bottom of the Unit 2 outside-containment junction box enclosures for the Viking Electrical Penetrations. Also, the electrical penetration inspection procedure is being amended to inspect the weep holes for blockage.	December 2011

No.	UNIT 3 HIGH ENERGY LINE BREAK LAR COMMITMENTS – 40H THROUGH 45H	DUE DATE
40H	The inlet isolation valves to the Letdown Coolers on the Letdown Line (3HP-1 and 3HP-2) will be upgraded to permit their use following a postulated HELB on the Letdown Line at Containment Penetration No. 6. With these valves upgraded, either could then be closed if either of the inboard containment isolation valves (3HP-3 and 3HP-4) fails to close in order to mitigate the postulated HELB on the Letdown Line.	To be provided to the Staff upon issuance of the SER.
41H	The Unit 3 Auxiliary Building HVAC ducting near the Unit 3 Control Complex is being upgraded with duct registers or cover plates to prevent the potential propagation of the HELB generated environment in the East Penetration Room to the Unit 3 Control Complex.	To be provided to the Staff upon issuance of the SER.
42H	The valves (3HP-103 and 3HP-107) on the individual suction lines to the "A" and "B" High Pressure Injection (HPI) pumps are being upgraded to allow the remote operation (operated outside the HPI pump room) of these valves. The remote operation of these valves allow the isolation of postulated HELBs on the discharge side of the HPI Pumps without compromising the availability of the other HPI Pumps and the need for maintaining the Letdown Storage Tank aligned to the HPI Pump suction piping. For a single active failure of either valve 3HP-103 or 3HP-107 to close, a redundant, remotely operated valve is provided on each of the HPI Pumps "A" and "B" to assure HELB mitigation.	To be provided to the Staff upon issuance of the SER.
43H	The position of the Unit 3 Plant Heating System isolation valve 3AS-182 being changed from "OPEN" to "CLOSED." This position change will eliminate the need to postulate Plant Heating System HELBs in the East Penetration Room and West Penetration Room, because these piping lines will be isolated during Normal Plant Conditions of the station.	To be provided to the Staff upon issuance of the SER.

No.	UNIT 3 HIGH ENERGY LINE BREAK LAR COMMITMENTS – 40H THROUGH 45H	DUE DATE
44H	Turbine Building structural support columns M-20 (Unit 1), M-35 (Unit 2), D-43 and D-45 (Unit 3), M-49 (Unit 3), and L-47 (Unit 3) will be modified by adding a brace or reinforcement to each column. These modifications are necessary to prevent potential failure of the column(s), when subjected to a pipe whip load.	To be provided to the Staff upon issuance of the SER.
45H	Weep holes will be installed in the bottom of the Unit 3 outside-containment junction box enclosures for the Viking Electrical Penetrations. Also, the electrical penetration inspection procedure is being amended to inspect the weep holes for blockage.	December 2011

No.	TORNADO LAR COMMITMENTS - 15T THROUGH 19T	DUE DATE
15T	Analyze the double column set which support each unit's Main Steam lines outside of the containment building, and provide modifications, as necessary, to meet tornado criteria	Complete
16T	Physically protect the Atmospheric Dump Valves (ADVs) per UFSAR Class 1 tornado criteria.	To be provided after completion of the SSF/MS line safety analysis
17T	Improve protection of the Standby Shutdown Facility (SSF) double doors (large 8'x12' doors located on the south side of the SSF structure) per UFSAR SSF tornado criteria.	12-2011
18T	Revise and clarify the tornado LB description as documented in UFSAR Section 3.2.2; add the TORMIS methodology results to UFSAR Section 3.5.1.3, and correct inaccurate tornado design information for the Auxiliary Building Cable and Electrical Equipment Rooms as described in UFSAR Table 3-23.	~~12-2010~~ 12-2011
19T	The SSF BASES for TS 3.10.1 will be clarified to address degradation of passive civil features as not applying to operability under Technical Specifications Limiting Condition for Operation (TS LCO) 3.10.1, "Standby Shutdown Facility," but rather as UFSAR commitments outside of the ONS TS.	~~12-2010~~ 12-2011

RAI 60 [T/H]

If the SSF or PWS systems are activated to mitigate a tornado or HELB what is the potential for accelerated corrosion of the steam generator tube which could lead to a steam generator tube rupture.

Duke Energy Response

If the SSF or PSW systems are activated to mitigate a tornado or HELB, lake water could be introduced to the steam generators. The use of lake water for these events does not constitute a change to the licensing basis for Oconee. In the original tornado and HELB analysis, the low pressure auxiliary service water system was the credited means of establishing feedwater to the steam generators. In addition, the low pressure auxiliary service water system was credited for long term decay heat removal following a loss of all external water sources. The PSW system is replacing the existing low pressure auxiliary service water system.

It is further noted that the steam generator tubes and key steam generator support structures (support plates, tie rods) are constructed of corrosion resistant materials. Specifically, the steam generator tubes are constructed of Alloy 690 material. The support plates and tie rods are constructed of stainless steel material.

The corrosion-resistant materials of construction of the steam generators should prevent accelerated corrosion of steam generator components that might result in tube rupture from such an event.

The following information includes a follow-up inquiry from the Staff after issuance of the October 8, 2010, RAI letter (denoted as an "RAI") in addition to proposed changes to the HELB and Tornado LARs by Duke Energy to correct inconsistencies in the original LAR submittals (denoted as an "Item").

RAI 61 [T]

In reference to the May 6, 2010, "Responses to Request for Additional Information for the License Amendment Request to Revise Portions of the Updated Final Safety Analysis Report Related to the Tornado Licensing Basis," additional clarification is requested. Please provide the basis for the following statements from the earlier responses:

Page 4, 4th paragraph

"There are several (described below) that affect assumptions in the TORMIS analysis."

Unclear what "assumptions" are being affected.

Page 9, 1st paragraph

"Significant damage is defined as damage that would prevent meeting a design basis safety function."

Page 9, 4th paragraph

"In some cases, multiple SSCs must be simultaneously damaged to be "important" and the TORMIS code can provided the probability of these simultaneous multiple strikes."

Page 10, 1st paragraph

Collectively, this list of items represents the frequency of damage to any individual SSC that could fail the Standby Shutdown Facility (SSF) mitigation strategy as described in the LAR.

Despite the reference to "multiple SSCs" on Page 9, 4th paragraph, it appears that only Tornado strikes on single SSCs whose failure would individually and directly prevent a design basis function are included in the evaluation. Please confirm whether this is accurate and discuss why multiple strikes are not considered.

Duke Energy Response

The response to RAI 2-2 (part b) in the May 6, 2010, letter discusses a set of plant modifications that Duke committed to implement to improve tornado missile protection. The physical location, dimensions, and material properties of plant structures and safety targets are key assumptions in the TORMIS analysis model. These modifications represent changes to the material properties of existing structures and new structures that have been incorporated in the TORMIS models. The installation of MSIVs provides an isolation function which eliminates main steam piping in the Turbine Building (downstream) from being considered as targets.

Situations in which multiple SSCs must be simultaneously damaged to affect the required plant safety function are addressed in Section 5.2 in Attachment 4 of the Oconee Tornado LAR. The specific cases that were identified were determined to have a negligible probability and were therefore omitted from the Table 5 results.

It is noted that the TORMIS code is capable of determining the probability of 2 targets at the same time but not 3 targets at the same time, and also will not estimate a damage probability of a target located below grade inside of a structure. These limitations affected the evaluation of two of the special cases. In the case of the "pedestal" columns, a sensitivity run was made for each unit (3 pairs of columns). The results of the sensitivity evaluations returned a zero probability of a simultaneous damaging strike on both columns.

Item 62 [T/H]

Duke Energy proposes to change the terminology "KHU underground path" given in several previously submitted TS, TSB, and UFSAR marked-up pages, to "KHU Protected Service Water Power Path." This change is being made to alleviate potential operator confusion since the pathway terminology used from the KHUs to the PSW switchgear building is not the same as the underground path from the KHUs to the CT4 blockhouse.

Duke Energy Action

The affected LAR pages affected are:

1. HELB LAR: PSW TS page 3.7.10-1
2. HELB LAR: PSW TS page 3.7.10-2 (SR 3.7.10.2)
3. HELB LAR: PSW TSB page B 3.7.10-4
4. Tornado LAR: Enclosure 2, page 9
5. Tornado LAR: Enclosure 2, page 14
6. Tornado LAR: Enclosure 2, page 23

These pages have been revised and the markups included in the Attachment to this submittal.

Item 63 [T]

The wording in the tornado LAR which states that all of the PSW ductbank is located underground requires revision. As such, Duke Energy proposes the following change:

Duke Energy Action

As stated in Enclosure 2 of the Tornado LAR [Ref. 7] beginning at the end of page 9 onto page10, *"As added margin, alternate power (primary power is from the KHU underground feed) to the new PSW System is provided from the Central Tie Switchyard via a 100 kV transmission line to a 100/13.8 kV substation located adjacent to the station and then via a 13.8 kV overhead path where it enters an underground ductbank leading to the PSW switchgear building."* This statement was factual at the time of Tornado LAR submission; however, as the design of the new ductbank has progressed, it has become necessary to locate limited portions of the ductbank above ground due to constructability and interference issues. The two areas where the new ductbank is above ground are where the new ductbank joins to the existing Keowee Underground Path and where the new ductbank crosses the Radwaste trench near the PSW building. The sections of the ductbank that are above ground have been designed to be tornado protected. Therefore, the entire PSW ductbank remains fully protected from tornado effects

Consequently, Duke Energy proposes to change the word "underground" to "tornado protected" in the quoted statement. The statement will now read, *"As added margin, alternate power (primary power is from the KHU underground feed) to the new PSW System is provided from the Central Tie Switchyard via a 100 kV transmission line to a 100/13.8 kV substation located adjacent to the station and then via a 13.8 kV overhead path where it enters a tornado protected ductbank leading to the PSW switchgear building."*

Attachment

Documentation for Responses to RAI 13, RAI 41, and Duke Energy Item 62

Response Documentation for:

RAI 13

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.1.a	Separation of Essential Systems for Pipe Ruptures.	Conforms	Redundant Systems necessary to achieve safe shutdown are physically separated[1].
B.1.b (entire section)	Break and Crack Exclusion in Fluid Piping located in Containment Penetration Areas (between isolation valves).	Optional, not applicable	Terminal End Breaks in Containment Penetration Areas originally postulated in the 1973 MDS report will be retained. Exclusion of break or critical locations is not used. Breaks or critical cracks in high energy lines located in the Containment Penetration Areas are based on stress thresholds
B.1.c.(1) (entire section)	Postulation of Pipe Breaks in Class 1 Piping Outside of the Containment Penetration Area.	Not Applicable	There is no Class 1 Piping located outside containment.

[1] The overall mitigation strategy is predicated on separation of essential systems (e.g., those systems and components necessary to reach safe shutdown) from the postulated high energy line break. For breaks postulated to occur in the Turbine Building, systems and components located in the Auxiliary Building or the Standby Shutdown Facility (SSF) would be available for mitigation of the effects from the break. For breaks postulated to occur in the Auxiliary Building, systems and components located in the Turbine Building or SSF would be available for mitigation of the effects from the break.

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.1.c.(2)(a)	Postulation of Pipe Breaks at Terminal Ends for ASME Class 2 or Class 3 piping Outside the Containment Penetration Area.	Partially Conforms	Terminal end breaks are not postulated at closed isolation valves separating a high energy system from a non high energy system, if the line containing the isolation valve is included in the stress analysis of the system and the stress analysis is continuous across the valves, such that valid stress information is available. Terminal end breaks are postulated at closed isolation valves separating a high energy system from a non high energy system for those cases when the system under consideration does not have a seismic stress analysis.
B.1.c.(2)(b)(i) **or** (ii)	For ASME Class 2 or 3 Piping: (i) Postulation of Intermediate Pipe Breaks at each pipe fitting **or** (ii) Postulation of Intermediate Pipe Breaks based on high stress.	Conforms	USAS B31.1. is the code of record for Oconee. Only SSF-Auxiliary Service Water piping meets classification as ASME Class 3 piping. SSF-ASW piping is classified as not operating during normal operations.

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.1.c.(3)	Intermediate Pipe Breaks for Non ASME seismically analyzed piping.	Partially Conforms	For seismically analyzed piping, breaks to be postulated in high energy systems at locations were primary + secondary stress equals or exceeds .8 x (Sa + Sh), determined in accordance with USAS B31.1 (1967 Edition). For non analyzed piping or analyzed piping that does not contain seismic loading, pipe ruptures are postulated at all pipe girth welds.

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.1.c.(4)	Design of structure(s) separating high energy lines from essential systems and components.	Conforms	Structure(s) separating high energy lines from essential systems and components are designed for the consequences of pipe break(s) as follows: For HELBs located in the Turbine Building, no pressurization loading is assumed on the structure due to numerous openings in the building, and the large open volume of the building that would prevent significant pressurization type loading. For HELBs located in the East Penetration Room of the Auxiliary Building, pressure relief features have been incorporated in the design of the building to prevent significant pressurization type loading. For HELBs located in other closed compartments in the Auxiliary Building, analysis has demonstrated that significant pressurization of the compartment will not occur. (See footnote 1)

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.1.c.(5)	Environmental Qualification of safety related equipment per SRP 3.11.	Partially Conforms	Equipment necessary to reach safe shutdown is environmentally qualified to the Environmental Qualification Criteria Manual. Oconee's response to IEB 79-01.b defines the Environmental Qualification program. (See footnote 1)
B.1.d	Identification of piping runs that contain postulated pipe ruptures required by B.1.c.	Conforms	High Energy systems have been identified and documented in Oconee plant calculations. Break locations within these systems likewise have been identified and documented in Oconee plant calculations.[2]
B.1.e.(1)	Postulation of critical cracks for ASME Class 1 piping based on stress located Outside Containment Penetration Areas.	Not Applicable	No Class 1 Piping outside containment.

[2] High Energy systems are identified in calculation OSC 8385. Break locations are identified in calculations OSC-7516.01 (Unit 1), OSC-7517.01 (Unit 2), and OSC 7518.01 (Unit 3).

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.1.e.(2)	Postulation of critical cracks for ASME Class 2 & 3 piping (and non ASME Class 1, 2, or 3) based on stress for piping located Outside Containment Penetration Areas.	Partially Conforms	For seismically analyzed piping, critical cracks are to be postulated at locations were primary + secondary stress equals or exceeds .4 x (Sa + Sh), determined in accordance with USAS B31.1 (1967 Edition).
B.1.e.(3)	Postulation of critical cracks in non safety class, non-analyzed piping located Outside Containment Penetration Areas.	Does not conform	Critical cracks are not postulated for non analyzed piping since the effects from postulated pipe breaks would bound the effects from critical cracks. Pipe ruptures are postulated for non-analyzed piping at all pipe girth welds. (See response to B.1.c.(3))
B.2.a	Separation of Essential Systems for Critical Cracks in Moderate Energy Systems.	Does not conform	Oconee not designed as a moderate energy plant.
B.2.b	Postulation of Critical Cracks in Containment Penetration Areas for Moderate Energy Systems.	Does not conform	Oconee not designed as a moderate energy plant. See response to Section B.1.b.
B.2.c.(1)(a)	Exemption to postulation of critical cracks for Moderate Energy Systems outside of the containment penetration areas.	Does not conform	Oconee not designed as a moderate energy plant.

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.2.c.(1)(b)	Postulation of critical cracks for moderate energy ASME Class 1 piping outside of the containment penetration areas based on stress analysis.	Not Applicable	No moderate energy Class 1 piping outside containment.
B.2.c.(1)(c)	Postulation of critical cracks for moderate energy ASME Class 2 or 3 piping outside of the containment penetration areas based on stress analysis.	Does not conform	Oconee not designed as a moderate energy plant.
B.2.c.(2)	Postulation of critical cracks for moderate energy piping not exempted by B.2.c.(1)	Does not conform	Oconee not designed as a moderate energy plant.
B.2.c.(3)	Postulation of critical cracks for non seismic moderate energy piping.	Does not conform	Oconee not designed as a moderate energy plant.
B.2.d	Postulation of critical cracks in moderate energy systems in proximity to high energy systems.	Does not conform	Oconee not designed as a moderate energy plant.
B.2.e	Postulation of critical cracks in piping systems that qualifies as high energy systems for short operational periods.	Does not conform	Oconee not designed as a moderate energy plant. Certain piping systems are high energy for short operational periods. For those systems, critical cracks are not postulated.

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.3.a.(1)	Postulation of circumferential breaks. Exemption of postulation of circumferential breaks for piping exceeding 1"nominal pipe size, except when hoop stress exceeds longitudinal stress by a factor of 1.5. Requirement that instrument lines meet Reg. Guide 1.11	Partially Conforms	Circumferential breaks postulated at locations determined in accordance with the response to items B.1.b, B.1.c.(2)(a), & B.1.c.(3). Exemption for postulation of circumferential breaks when actual stress greater than break threshold, but the hoop stress exceeds the longitudinal stress by a factor of 1.5 not taken. Instrument lines do not meet Reg. Guide 1.11
B.3.a.(2)	Postulation of circumferential breaks for non analyzed lines	Conforms	
B.3.a.(3)	Circumferential breaks should result in complete severance and separation amounting to at least one diameter lateral displacement.	Conforms	
B.3.a.(4)	Dynamic force of the jet from circumferential breaks should be based on effective flow area and on a calculated fluid pressure as modified by a thrust coefficient. Obstructions to flow or absence of energy reservoirs may be taken into account for reduction of the jet discharge.	Conforms	Dynamic forces determined in accordance with ANS 58.2 (1988).

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.3.a.(5)	Pipe whip from circumferential breaks should occur in the plane defined by the piping geometry and pipe movement should be in the direction of the jet reaction.	Conforms	
B.3.b.(1)	Postulation of longitudinal breaks. Exemption of postulation of longitudinal breaks for piping equal to and exceeding 4"nominal pipe size, when hoop stress exceeds longitudinal stress by a factor of 1.5.	Conforms	Exemption not taken.
B.3.b.(2)	Longitudinal breaks need not be postulated at terminal ends.	Conforms	
B.3.b.(3)	Orientation of longitudinal breaks.	Conforms	
B.3.b.(4)	Dynamic force of the jet from longitudinal breaks should be based on effective flow area and on a calculated fluid pressure as modified by a thrust coefficient. Obstructions to flow or absence of energy reservoirs may be taken into account for reduction of the jet discharge.	Conforms	Dynamic forces determined in accordance with ANS 58.2 (1988).
B.3.b.(5)	Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural restraints or piping stiffness.	Conforms	

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.3.c.	Leakage cracks should be postulated at those locations specified in B.1.e for high energy piping, and B.2.c.(1) for moderate energy piping.	Partially Conforms	Leakage cracks postulated for high energy piping, but not for moderate energy piping. Oconee is not designed as a moderate energy plant.
B.3.c.(1)	Leakage cracks need not be postulated in piping 1" and smaller.	Conforms	
B.3.c.(2)	For high energy systems, leakage cracks should be postulated at circumferential locations that result in the most severe environmental conditions. For moderate energy systems, see B.2.c.(2)	Partially Conforms	Leakage cracks postulated for high energy piping at locations postulated in accordance with response to B.1.e.(2). No cracks are to be postulated for non-safety piping locations in accordance with response to B.1.e.(3).
B.3.c.(3)	Fluid flow from a leakage crack shall be based on circular opening of area equal to ½ the piping diameter by ½ the piping wall thickness.	Conforms	

Table RAI-13: Comparison of Oconee HELB Design Basis with BTP MEB 3-1 (Revision 2)

BTP MEB 3-1 Section	Description	Oconee Design	Comments/Clarifications
B.3.c.(4)	Flow from a leakage crack should be assumed to result in an environment that wets all unprotected components within the compartment and communicating compartments. Flooding should also be assumed for the compartment containing the leakage crack and those communicating compartments. Flooding effects should be based on a conservative estimate of the time period required to effect corrective actions.	Partially Conforms	Universal wetting of all components necessary to achieve cold shutdown from leakage cracks located within closed compartments are not assumed. 'Direction wetting' associated with jet impingement will be assumed and components necessary to achieve cold shutdown will be protected from 'directional wetting.' Flooding effects are considered as appropriate.

Response Documentation for:

RAI 41

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
C1	**Design Considerations for PSW/HPI** a. "...the commitment should specify that the PSW/HPI and related switchgear modifications will satisfy safety-related, seismic Category 1 criteria, and will be controlled and maintained in accordance with 10 CFR 50, Appendix B criteria." "The 'PSW System would be designed and constructed to meet Duke's standards for a safety-related system (QA-1).' Why isn't this characterized as a commitment (see Page 10 of Attachment 3, "Regulatory Commitment Table," fifth bullet)?" "While the licensee seems to suggest that the PSW/HPI system will be installed as safety-related, seismic Category 1, and will be controlled in accordance with 10 CFR 50 Appendix B requirements, this needs to be clearly stated to assure that there is no misunderstanding." b. "Why aren't PSW/HPI design criteria and time critical actions included similar to HELB commitments that were made?" "...why aren't these PSW/HPI design considerations reflected in the tornado commitments?"	a. The intent was to include the PSW/HPI System and the East Penetration Room flood prevention modifications to be designed and constructed to meet Duke's standards for a safety-related system (QA-1) per the Duke Quality Assurance Program Topical Report and described as such on the LAR. b. This was simply an attempt to reduce duplication within the letter.	a. Agreed - Common Understanding – No Further Action Required b. Agreed - Common Understanding – No Further Action Required
C2	**GL 91-18 Actions** a. "While HELB and tornado mitigation strategies are being implemented, any future issues that are identified as a result of these activities will be entered into Oconee Nuclear Station (ONS) corrective action program - no mention of GL 91-18 actions to address issues of this nature, or other actions that will be taken to assure that NRC requirements are satisfied."	a. The Duke Corrective Action Program requires items entered into the corrective action program to be evaluated for applicability of Operability and actions needed to address compliance with NRC requirements (NRC Inspection Manual Part 9900).	a. Agreed - Common Understanding – No Further Action Required
C3	**SSF Risk Reduction Effort** a. "In parallel with this, a risk reduction effort has been initiated that is intended to improve the reliability and availability of the standby shutdown facility (SSF) - there was no mention of a commitment or follow up with the NRC for this item."	a. The SSF risk reduction effort was initiated in 2005 in order to improve the reliability and availability of the SSF independent of resolution of tornado and HELB licensing basis issues.	a. Agreed - Common Understanding – No Further Action Required

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H1	**<u>Volumetric Inspections of Piping in lieu of Protection of Equipment</u>** "In Attachment 4 to the November 2006 letter, Duke proposes to use periodic volumetric examinations in lieu of evaluating the effects of pipe rupture at most of the pipe rupture locations in the turbine and auxiliary buildings. The proposed alternative to use periodic volumetric examinations in lieu of pipe rupture evaluation is not part of the criteria contained in the Giambusso letter or the criteria contained in BTP MEB 3-1. BTP MEB 3-1 requires 100% volumetric examination of all welded connections between the containment isolation valves in addition to meeting the stress limits specified in B.1.b of the BTP MEB 3-1. The basis for the BTP MEB 3-1 criteria is to provide a high level of assurance that breaks do not occur in the critical area between the containment isolation valves. BTP MEB 3-1 does not contain a provision for performing periodic volumetric examinations as an alternative to postulating the pipe cracks and ruptures at the locations required by BTP MEB 3-1."	Oconee proposes that for those postulated break or crack locations that have the potential to affect systems and components necessary to reach safe shutdown, including those that could affect the main steam pressure boundary, and those locations that have the potential to affect the turbine building structure, periodic volumetric inspections would be instituted in lieu of providing protection (e.g. pipe whip restraints, jet shields, etc.). While the exact number of inspection locations is uncertain at this time, it is generally believed to be less than 50 locations per unit. More than four thousand break locations per unit have been evaluated, so the characterization of 'most' is inaccurate. Oconee believes that detection and prevention of a postulated break location is superior to providing physical protection. Further, such structural modifications would (1) not provide a risk benefit, (2) would hamper normal plant maintenance activities, and (3) limit inspection access to the very location(s) where the break(s) are postulated. There is some precedent in this area. Another B&W unit, similar to Oconee, has incorporated a similar inspection program into their technical specifications, although not to the scale proposed by Oconee. Finally, the proposed program is a logical extension of the in-service inspection plan, where periodic inspections are used to demonstrate the structural integrity of safety related piping.	Agreed - Open Issue. *[No credit is taken for these inspections with regards to the identification of postulated HELBs in the Turbine Building and Auxiliary Building.]*
H2	**<u>BTP MEB 3-1 USE</u>** "Revision 2 to BTP MEB 3-1 also contains additional criteria not provided in the Giambusso letter. The staff has repeatedly requested Duke to compare its proposed HELB criteria with the full criteria contained in BTP MEB 3-1 in order for the staff to perform a thorough safety review of the Duke HELB proposal. The November 30, 2006, letter only addresses the criteria from BTP MEB 3-1 which provide relaxations to the Oconee licensing basis HELB criteria." "In order for the staff to perform this licensing amendment review, it will be necessary for Duke to clearly address how its proposed new licensing basis meets all the criteria in BTP MEB 3-1 or provide a basis for any deviations to the criteria. While most of the specific commitments proposed by Duke in the November 30, 2006, letter are considered to be acceptable, the staff does not fully agree with those that relate to the specific issues identified below." "Duke needs to provide a specific justification for each pipe rupture location it plans to deviate from the staff guidance in BTP MEB 3-1."	Oconee's HELB design basis will continue to be the Giambusso/Schwencer letters, as amended by GL 87-11 and our letter dated 11/30/06. Oconee does not plan to adopt MEB 3-1 except as noted below. GL 87-11 notes that "Licensees of operating plants desiring to eliminate previously required effects from arbitrary intermediate pipe ruptures may do so without prior NRC approval, unless such changes conflict with the license or technical specifications." Oconee believes no further justification is needed for the adoption of GL 87-11, beyond that prescribed by the GL. Other facilities have adopted GL 87-11 in a similar fashion. The 11/30/06 letter describes the use of MEB 3-1 on two occasions: (1) For piping that is not analyzed or does not include seismic loadings; intermediate breaks will be postulated as provided in MEB 3-1. This means that breaks will be postulated at all girth weld locations irrespective of the stresses in the pipe. This clearly is not a deviation from Giambusso/Schwencer, which stipulates that a minimum of two breaks per run be postulated. This approach is more conservative than Giambusso/Schwencer. (2) For equivalent Class 2 and 3 piping that is seismically analyzed, critical cracks will be postulated at axial locations where the calculated stress for the applicable load case exceed .4(Sa + Sh). This is a deviation from Giambusso/Schwencer, which stipulates that critical cracks be postulated at the most adverse location independent of stress. However, the 11/30/06 letter justifies this by noting that "Adoption of this provision will allow the station to focus attention to those medium and high stress areas that have a higher potential for leakage cracks to form."	Agreed - Common Understanding - additional information to be provided in the LAR. *[Information on the use of criteria in BTP MEB 3-1 is provided. The criteria, upon which the HELB Break locations and types are determined, are identified in Section 2.0 of the HELB Report. The listed criteria identify how and where break locations and types are chosen.]*

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H3	**Definition of High Energy System per Footnote 5 of MEB 3-1** "...the Duke letter does not indicate whether its proposal fully satisfies the position in footnote 5 of BTP MEB 3-1, Revision 2. Specifically, footnote 5 states that systems operated during PWR startup, hot standby, or shutdown qualify as high energy systems. Duke needs to clarify that it will satisfy the definition of high energy system contained in footnote 5 of BTP MEB 3-1."	Oconee has no plans to adopt any other portions of MEB 3-1, including footnote 5. As regards footnote 5, Oconee plans to eliminate systems that operate for short periods of time at high energy conditions due to the low probability of a break occurring during high energy operations. We previously communicated that we would provide historical information regarding system operating times in the LAR(s).	Common Understanding - Additional information to be provided in the LAR. (Reference NRC letter dated 7/12/2006, Enclosure 2, Item 18) *[The definition of a high energy system is provided in Section 1.5 of the HELB Report.]*
H4	**Postulation of Terminal End High Energy Line Breaks at Closed Ended Valves** "The Duke letter does not indicate whether its proposal fully satisfies the position in footnote 3 of BTP MEB 3-1. Specifically, footnote 3 states that for piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve. This means that a pipe rupture would have to be postulated at the connection to the closed valve. Duke needs to clarify that it will satisfy the complete criteria contained in footnote 3 of BTP MEB 3-1."	Although not addressed by Giambusso/Schwencer, Oconee intends to postulate breaks/cracks at closed valves as follows: The postulation of terminal end breaks at the first normally closed valve(s) separating portions of a system maintained pressurized during normal operations and portions of a system not maintained pressurized depends on whether the system has a seismic analysis that is continuous across the valve. For system or portions of systems that are not seismically analyzed, breaks are postulated to occur at all piping girth welds in the system including those that attach to normally closed valves. For systems or portions of systems that are seismically analyzed, and the analysis is continuous across the normally closed valve, such that stresses can be accurately determined, break and crack locations are determined based on comparison to the break and crack stress thresholds. This interpretation for boundary valves in seismically analyzed lines has been previously approved by the staff for the LPI cross tie submittal (Oconee), for the revised pipe rupture analysis criteria (Crystal River), and for the "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" (Watts Bar).	Agreed - Common understanding - No additional action
H5	**Treatment of the Letdown Line as a High Energy Line** "In Attachment 5 to the November 2006 letter, Duke argues that the reactor coolant letdown line outside the containment does not qualify as a high energy system in accordance with its licensing basis because the system does not exceed both 200 degrees F and 275 psig. However, the Oconee licensing basis criteria provided in Duke Report No. OS-73.2, "Analysis of the Effects Resulting from Postulated Piping Breaks Outside Containment for Oconee Nuclear Station, Units 1, 2, and 3, " clearly states that a higher energy lines are defined as those that have either a normal service temperature greater than 200 degrees or a pressure greater than 275 psig. This is the same criterion that is referenced in BTP MEB 3-1. Duke needs to treat the reactor coolant letdown line as a high energy line up to the isolation valve."	Oconee agrees that the Letdown line should be considered as high energy. However, upon rereading the original SER for Oconee, there seems to be some confusion on this point. The SER clearly notes the following, "The reactor coolant letdown is cooled before leaving the reactor building so this system is essentially a high pressure system rather that a high pressure and temperature system." Although not explicitly stated in the SER, it is believed by Oconee that this statement allowed some latitude in the postulation of single active failures, as follows:	Agreed - Common understanding - No additional action

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H6	**Single Active Failure Criterion for the Letdown Line Break between the Containment Penetration and the Outboard Containment Isolation Valve** "...the NRC staff does not agree with the licensee's characterization in this commitment of the plant licensing basis relative to the letdown line; the single failure criterion is applicable and must be considered..." "Contrary to Duke's position, the MDS Report (Section 3.1.9) indicates that the break is isolated by automatic closure of xHP-3, xHP-4, and xHP-5; and Duke did not take exception to the single failure criterion for this break scenario. In fact, for those break locations where the MDS report did find that the single failure criterion was not satisfied, the condition was specifically recognized and interim compensatory measures and plant modifications were identified for resolving the single failure discrepancies that were found. Furthermore, Duke indicated that the NRC criteria that were specified for addressing HELB were satisfied. Therefore, the plant licensing basis for postulated failures of the letdown line includes consideration of single active failures." "It is the NRC staff's position that the plant licensing basis for postulated failures of the letdown line includes consideration of single active failures, and postulated failures of the letdown lines for the Oconee units must be addressed accordingly."	It is clear that no single active failure was postulated in the original MDS report. The report noted that valves HP-3, 4, & 5 could be used to isolate the break. However, HP-3 and 4 are located in parallel lines downstream of their respective Letdown Coolers, upstream of the break location. HP-5 is located outside containment, downstream of the postulated break location. So, a failure of HP-3 or 4 to close would result in an un-isolated break. Closing HP-5 was and is not important, since it is downstream of the break location. However, HP-1 & 2 can be closed, by manual operator action inside the control room to isolate the break. Oconee will include, as part of the LAR(s), a description of the dose re-analysis of this break scenario, crediting closure of HP-1 & 2 to isolate the break.	Agreed - Common Understanding - additional information to be provided in the LAR. *[This information is provided in Sections 4.0 - 6.0 for each unit, described in Section 7.3, and the modification to xHP-l and xHP-2 is described in Section 9.0. No further action on this item is required.]*
H7	**Location of Terminal End Breaks at Small Bore Reactor Building Penetrations** "In Attachment 4 to the November 2006 letter, Duke indicates that breaks will not be postulated at the penetration anchors for small bore piping penetrations because the penetration anchors are located inside the containment. Instead, Duke indicates that breaks are postulated in the piping run outside the containment wall and remote from the anchor. This is not consistent with the criteria provided in Section 2.1 of Duke Report No. OS-73.2 which requires break locations at the terminal end of the piping run. BTP MEB 3-1 also requires postulation of breaks at the terminal end. The basis for the criteria is that breaks are expected to occur at locations that provide rigid constraint to the piping, such as anchor points. Duke needs to either evaluate the effect of pipe breaks at the terminal end (anchor point) as required by the criteria or provide justification as to why the alternative location it selected is the most likely location for a HELB."	The MDS report provided drawings of break locations at the small bore Reactor Building penetrations. These locations were clearly inside the penetration rooms, beyond the piping to reactor building liner welds. In addition, the aforementioned SER provides the following: "The staff agrees with the applicant's selection of pipe failure locations and concludes that all required accident situations have been addressed appropriately by the applicant." The consequences of the small bore break locations at the RB penetrations documented in the MDS report would be very similar as to those postulated at the piping to liner welds, except in one respect, their affect on containment integrity. However, other analyses evaluated the affect on containment should a break occur at the pipe to liner weld, as part of the containment design. The design basis for these analyses is described in Section 3.6.1.1 of the UFSAR. The design basis is as follows: (1) All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures. (2) All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures. (3) All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity. (4) All secondary lines whose breaks could damage a primary line and also breach containment are designed to maintain containment integrity. In conclusion, Oconee does not believe that it is necessary to change the licensing basis for postulation of breaks at small bore penetrations.	Agreed - Common Understanding - additional information to be provided in the LAR. *[This information is not explicitly addressed in the HELB Report, although all small bore postulated breaks at the Reactor Building Penetrations have been analyzed and described. Information on the exact point of the break on the line is not provided. However, since this appears to apply only to the Letdown Line and the RCP Seal Injection Lines, further information is not necessary. The effects of these breaks are addressed in the HELB Report, sections 4.2, 5.2, & 6.2.]*

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H8	**Location of Terminal End Break at the Main Steam Reactor Building Penetration** "In Attachment 5 to the November 2006 letter, Duke indicates that breaks were not postulated at the east penetration room main steam terminal end anchor point because the penetration anchor is located inside the containment. Instead, Duke indicates that the break is postulated in the piping run outside the containment wall and remote from the anchor. This is not consistent with the criteria provided in Section 2.1 of Duke Report No. OS-73.2 which requires break locations at the terminal end of the piping system. Duke needs to either evaluate the effect of a pipe break at the terminal end (anchor point) as required by the criteria or provide justification as to why the alternative location it selected is the most likely location for a HELB."	The MDS report provided drawings of the two Main Steam break locations at the Reactor Building penetrations. These locations were clearly inside the penetration rooms, beyond the MS rupture restraints and the Reactor Building liner welds. As noted for the small bore breaks at the RB penetrations, the SER agreed with the selection by Oconee of the pipe failure locations and further concluded that all required accident situations had been appropriately addressed. The consequences of a MS break at the locations depicted in the MDS report would be very similar to those postulated at the MS rupture restraint. The rupture restraint, which forms part of the containment boundary, is connected to the MS piping by two welds. These welds connect the MS piping to a collar plate that is in turn welded to the rupture restraint. The inboard weld (RB Side) is designed such that should a break occur at the outboard weld (EPR side) containment integrity would not be threatened. Similarly, the outboard weld is designed such that should a break occur at the inboard weld containment integrity would not be threatened. The design of MS penetration and rupture restraint form part of the overall containment design. In conclusion, Oconee does not believe that it is necessary to change the licensing basis for postulation of breaks at the Main Steam penetrations.	Agreed - Common Understanding - additional information to be provided in the LAR. *[Pages 8-3 to 8-5 of the HELB Report only address the Feedwater Line restraint. Main Steam and small bore pipe penetrations [including drawings] are describe in Duke Energy's response to RAI 10 from the 10/23/2009 RAI response submittal.]*
H9	**Water Hammer Loads** "How are water hammer loads addressed?"	For those piping systems where water hammer is a concern (Main Steam & Main Feedwater), the calculation of Equation 9 (occasional loads) is based on the greater of OBE seismic or water hammer stresses.	Agreed - Common Understanding - Additional information to be provided in the LAR. *[The water hammer loads have been included in the Piping Analysis Calculations for the main feedwater, and the steam hammer loads have been included in the piping analysis calculations for the main steam piping. These piping analysis calculations are listed in the HELB Report tables in calculations OSC-7516.01 (unit 1), OSC-7517.01 (unit 2), OSC-7518.01 (unit 3.]*
H10	**Technical Specifications for PSW & SSF** "...licensing basis clarity should be reflected in the UFSAR, and TS requirements should be established in accordance with 10 CFR 50.36 requirements." "Operability of the water inventory for the SSF and PSW/HPI must be addressed. The current SSF TS in this regard was based on the availability of other systems such as EFW for performing the SSDHR function, which is not valid for the proposed tornado and TB HELB mitigation strategies. Furthermore, both the SSF and the PSW/HPI systems rely upon the same water supply and the licensee has not addressed how the water supply will be assured for both tornado and HELB mitigation." "Indicates that installation of PSW and HPI improvements will reduce reliance on the SSF by providing a system capable of independently establishing safe shutdown conditions, thereby significantly improving overall plant risk - not truly independent due to shared water source and west penetration room (WPR) vulnerabilities; no mention of establishing a Technical Specification (TS) requirement pursuant to 10 CFR 50.36 even though the	Oconee agrees that licensing basis for HELB will be reflected in the UFSAR. As documented in our 11/30/06 letter, the PSW has been evaluated regarding inclusion into the TS, and that evaluation concluded that the PSW operability requirements should be incorporated into the Selected Licensing Commitments (SLC) Manual and its Bases. This conclusion was based on 10CFR50.36 requirements, preliminary Oconee PRA results, and the applicability of NUREG 1430 for standard technical specifications. Regarding the assurance of the water supply for both SSF and PSW/HPI, see issue H12. The SSF will remain risk significant and its TS will remain as	Agreed - Common Understanding - Additional information to be provided in the LAR concerning a TS or SLC for only. Agreed - Common Understanding - Additional information to be provided in the LAR concerning RCS leakage. Agreed - Common Understanding - Additional information to be provided in the LAR concerning the submersible pump.. *[The Technical Specifications for the PSW system are provided in Attachment 2 of the Unit 1 HELB LAR. This information includes the surveillance requirements for the Submersible Pump. The issue of RCS leakage and SSF RC Make-Up Pump is not unique to the HELB Project, and it is not included in the HELB Report.]*

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
	licensee recognizes that the PSW/HPI modifications will "significantly" improve overall plant risk." A TS is required for the PSW/HPI system in accordance with 10 CFR 50.36(c) (2) (ii) (D). As stated on Page 3 of the November 30, 2006, submittal, Duke indicated that "the installation of a new PSW system and HPI system improvements will reduce reliance on the SSF by providing a system capable of independently establishing safe shutdown conditions, and thereby significantly improve overall plant risk." TS requirements were required for the SSF to assure its SSDHR function (even though other sources of SSDHR were considered to be available). The risk significance of the PSW/HPI system is on the same order of magnitude as the SSF and in this case, other sources of SSDHR may not be available. * Tornado and HELB events at Oconee represent at least the same level of risk as associated with design basis accidents (DBAs). * The licensee proposes to rely upon the PSW/HPI system in conjunction with the SSF for tornado and HELB mitigation, and the licensee's TORMIS analysis is predicated on this. Therefore, TS requirements should be established not only to assure the operability of the PSW/HPI system, but also to assure that both the SSF and PSW/HPI systems are not both rendered inoperable at the same time. *The PSW/HPI capability is the only means that can be relied upon for tornado and HELB mitigation beyond 72-hours, and it is the only means available for cooling down the Oconee units. "What limitations are required relative to reactor coolant system (RCS) leakage when using the MSRVs and atmospheric dump valves (ADVs) for steam generator (SG) pressure control and crediting the SSF, and what changes are necessary to the TS in this regard?" "The plant licensing basis for both tornado and HELB includes the capability to achieve and maintain cold shutdown conditions. In the case of tornado, the station ASW system is credited for being able to maintain SSD for at least 30 days and the same capability should be provided by the PSW/HPI system. The submittal needs to explain how this *capability will be assured, especially with respect to TS requirements."* "TS requirements that assure the operability and availability of structures, systems and components (SSCs) that are relied upon for the tornado and HELB mitigation strategies must be established, such as for the standby shutdown facility (SSF), the PSW/HPI system, the Unit 2 condenser circulating water (CCW) system, and for reactor coolant system leakage." "The current SSF TS requirements did not include consideration of the proposed mitigation strategy and the current 45 day allowed outage time (AOT) for the Unit 2 CCW inlet is of concern. This needs to be reconsidered since the basis for the 45 day AOT is no longer valid, and the AOT should be limited based on tornado and HELB considerations recognizing that there are not other sources of water." "Page 3, first bullet: Relative to the capability to power the submersible pump by the PSW switchgear, what TS operability and surveillance requirements are appropriate?"	currently written. As noted in the 11/30/06 letter, the new PSW system will not mitigate any Oconee UFSAR Chapter 15 design basis events. Further, preliminary PRA indicates that the risk impact of PSW intended functions are lower than those of SSF. The addition of PSW/HPI actually reduces the safety significance of the SSF. Additionally, preliminary PRA analysis indicates that the AOT for the PSW/HPI system would be ~21 days as compared to the 7 day AOT of the SSF. Currently, the limiting condition for RCS leakage is maintained in accordance with TS 3.4.13 and the limiting condition for operation of the SSF is maintained in accordance with TS 3.10. The Commitment relative to operation of the Station ASW and SSF for the purpose of tornado mitigation is in accordance with SLC 16.9.9. The Maximum Allowed Total Combined RCS Leakage Rate was chosen to ensure that the seal leakage rate for all four (4) RC pumps plus other RCS leakage during normal operation remains low enough to allow the SSF RC Make Up System to maintain adequate inventory in the RCS to sustain natural circulation flow during an SSF event. The original Station ASW system, that also takes suction form the CCW header, is governed by SLCs, not TSs. The combined PSW/HPI and SSF tornado and HELB mitigation functions will be monitored using a revised version of SLC 16.9.9.	

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
		The limiting condition for the submersible pump is outlined in TS 3.10.1.b. The submersible pump provides long term makeup to the reservoir. The submersible pump is stored in the SSF facility. The surveillance requirements will remain the same. (See 1st bullet, Page 3 of Attachment 1 and Sections 1.1 and 1.2 of Attachment 2 in the November 30, 2006 letter)	
H11	<u>**72 Hour Mission Time of the SSF**</u> "The plant licensing basis is to be able to mitigate HELB events, including consideration of single active failures, and to place the plant in cold shutdown condition. The onus is on the licensee to demonstrate that the 72-hour mission time of the SSF is adequate for this purpose (e.g., extent of damage and time required to make necessary repairs and to resolve postulated failures of the PSW/HPI must be addressed)." "The 72-hour mission time of the SSF does not establish what the mission time is for mitigating HELB scenarios. Adequate assurance must be established that the PSW/HPI and SSF are capable of mitigating the HELB event to the point of establishing cold shutdown conditions, irrespective of the SSF mission time. The 30-day capability of the PSW/HPI system can be credited in this regard, but assurance that sufficient water inventory will be available and that the PSW/HPI can be restored within 72-hours is required."	The proposed HELB design basis is predicted on the ability to reach safe shutdown(1) within 72 hours. The SSF can adequately provide this function. Within the 72 hour timeframe, damage repair measures will be credited to insure the required systems and components are available such that an orderly progression to cold shutdown can begin. Oconee agrees that more detail should be provided (i.e. PSW single failure mitigation) on the scope and detail of these repair measures. Such detail and scope can be provided in the unit specific LAR(s). Regarding the availability of the water source to the SSF and PSW/HPI, there are no direct threats to the supply from postulated HELBs. With the use of the submersible pump discussed elsewhere in this presentation, the CCW can be replenished from the Lake Keowee source indefinitely. Again such detail can be provided in the LAR(s).	Agreed - Common Understanding - Additional information to be provided in the LAR. *[This information is provided in Section 3.3 of the HELB Report. Moreover, the statement is made that the PSW-ASW System will be restored to operability within 72 hours.]*
H12	<u>**Assurance of Suction Source**</u> "Contrary to the information that was provided, this PSW/HPI system is not totally independent of the standby shutdown facility because they share the same water source." Furthermore, both the SSF and the PSW/HPI systems rely upon the same water supply and the licensee has not addressed how the water supply will be assured for both tornado and HELB mitigation." "How is water supply from the Unit 2 CCW assured to be available? The existing TS AOT must be reconsidered accordingly recognizing the new tornado and HELB mitigation functions." "The licensing basis includes the capability to place the plant in cold shutdown and the mitigation strategy does not adequately address how this capability is assured relative to the extent of damage that can be experienced, recognizing that: a) it is critical to recover PSW/HPI within the 72-hour mission time of the SSF, and b) an assured source of cooling water that is good for at least 30-days is needed for the three Oconee units at the onset of	Oconee agrees that the water source for both SSF and PSW/HPI is not redundant. However, given that either system, but not both, will draw on this source, and given the available inventory, HELBs in the TB that result in the loss of 4160V, can be adequately mitigated such that safe shutdown(1) can be maintained for 72 hours following the event by use of the submersible pump. Following that period, there remains adequate CCW inventory to support an orderly cooldown to cold shutdown. Should the cooldown period exceed the capacity of the available inventory, the submersible pump can be used to refill the CCW from Lake Keowee. This activity can be easily achieved before depletion of the available inventory.	Agreed - Common Understanding - Additional information to be provided in the LAR. *[The PSW System and the SSF System share the same source of water for the PSW and SSF-ASW Subsystems. However, as stated in Sections 3.2 and 3.3 of the HELB Report the source is the water contained in the "Unit 2 CCW imbedded piping," which is a passive source. The report also states that the SSF is a backup to the PSW, which means that PSW and SSF will not be used simultaneously, and the submersible pump can be used to refill this source. All of this information is provided in Sections 3.2 & 3.3 and is used in the interaction analysis. No further actions or clarifications are required.]*

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
	tornado and HELB events." SSF and PSW will both use the Unit 2 condenser circulating water (CCW) inlet piping as a water source. How will availability of this water source be assured?		
H13	**Main Steam HELBS** "...the SSF cannot be credited as backup if the non-MS HELB results in a plant cooldown that exceeds SSF reactor coolant system (RCS) makeup capability, such as the turbine bypass valve (TBV) and feedwater control valve (FWCV) failures that are referred to on page 10 (for example). Also, it would seem that if this is a problem for non-MS HELBs, that it would be a problem for MS and main feedwater (MF) HELBs (also see Page 11, third paragraph)? Per Page 10, third paragraph, Duke to confirm the adequacy of previous analysis that the MS HELBs in the turbine building satisfies the specified criteria (no damage to protection systems, Class 1E electrical systems, or ES equipment on the affected unit, plus single failure consideration) such that the PSW/HPI and SSF do not have to be credited "The SSF cannot be credited for backup mitigation if the non-main steam (MS) HELB results in a plant cooldown that exceeds SSF RCS makeup capability (which appears to be the case for postulated turbine bypass valve (TBV) and feedwater control valve (FWCV) failures as referred to in Attachment 4, page 10, of the submittal (for example)."	Oconee recognizes that the SSF RCMU has limited capacity for RCS inventory control. As noted in Oconee's response to Information Notice 79-22 and reiterated in our letter dated 11/30/06, the profile considered for the environmental evaluation of the turbine bypass valve and feedwater control valve was based on a MS break. Oconee has no information, at this time, that indicates that these valves fail open for non-Main Steam breaks. As indicated in the 11/30/06 letter, work continues on the mitigation strategy for MSLBs and other HELBs that may result in a compromise of the MS pressure boundary. This analysis will consider non-safety control system malfunctions induced by environmental effects, the validity of the assumed environmental profile in the TB and the capabilities of the PSW/HPI system and the SSF to mitigate these HELBs.	Agreed - Common Understanding - Additional information to be provided in the LAR. See Item H1. *[These items are addressed in detail in Section 7.0 of the HELB Report. Moreover, Section 7.0 of the HELB report includes the analysis of the SSF/MSIV's for HELB event mitigation. No additional actions are necessary for these items.]*
H14	**HELB's and an Uncontrolled Blowdown of Either Steam Generator** "The consequences of HELB is determined based upon appropriate analyses, and the assumption that HELBs do not result in an uncontrolled blowdown of either SG (or excessive cooldown for that matter) must be justified accordingly, as well as any other assumptions that are credited in the HELB analyses. The HELB analyses must also address single failure considerations without exception." "The consequences of HELB are determined based upon appropriate analyses, and the assumption that HELBs do not result in an uncontrolled blowdown of either steam generator (SG), or excessive cooldown for that matter, must be justified accordingly (as well as any other assumptions that are credited in the HELB analyses)."	The unit specific LAR(s) will provide information and or references that demonstrate the consequences of all postulated HELBs. Information regarding a potential uncontrolled SG blowdown and the potential mitigation strategy will also be reported in the LAR, as appropriate. The postulation of single active failures will be addressed.	Agreed - Common Understanding - Additional information to be provided in the LAR. *[This information is provided in the break analysis in section 7.0 of the HELB Report and the individual break evaluations are provided in Sections 4.0, 5.0 & 6.0 of the HELB Report (ONDS-351).]*

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H15	**Control Room Cooling System & Main Steam Line HELBs in the Turbine Building** "...why isn't this sort of thing a problem for the MSLB in the TB (i.e., HELB in the TB can cause a loss of chilled water and power for HVAC; loss of colored buses)?"	As stated in the 11/30/06 letter, analysis has shown that the main CR would remain habitable and the equipment located there would there would remain functional for a prolonged loss of HVAC. The route of the Main Steam lines is not proximate to the CR. In addition, the TB is a large structure with numerous openings. As such, should a MSLB occur in the TB, the jet flow would be sufficiently far away from the CR such that the CR would continue to function, even with a loss of chilled water. Regarding the 4160V power, all direct interactions from HELBs postulated to occur in the TB are being evaluated, including interactions with the 4160V power.	Agreed - Common Understanding - Additional information to be provided in the LAR. (Reference NRC Letter dated July 12, 2006, Enclosure 2, Item 21) *[This item is addressed in Sections 3.8.1 and 8.0 (pages 8-29 and 8-30) of the HELB Report. Also, Sections 4.0 - 6.0 address the Turbine Building HELBs and their lack of interaction with the Auxiliary Building. No other action on this item is required.]*
H16	**Justification of 100% Humidity Non-Condensing** "The environmental profile is determined based upon analysis of the actual conditions that will exist following the pipe break, and the assumption that the environment is "non-condensing" must be justified and supported by the analysis." "The environmental profile is determined based upon analysis of the actual conditions that will exist following the pipe break, and the assumption that the environment is "non-condensing" must be justified and supported by the analysis."	As noted in Oconee's response to Information Notice 79-22 and reiterated in our letter dated 11/30/06, the profile considered for the environmental evaluation of the turbine bypass valve and feedwater control valve was based on a MS break. During normal operation, the SGs produce at least 60 degree superheated steam. Under such conditions, the amount of condensation is negligible.	Agreed - open issue Issue is really broader and concerns EQ. *[The Environmental Qualification Criteria Manual (EQCM) establishes the current licensing basis.]*
H17	**Restoration of PSW/HPI** "No flood protection will be provided for systems and components in the TB that are necessary to reach cold shutdown (CSD). This could require the plant to be maintained in safe shutdown (SSD) conditions for an extended period of time which places additional importance on the PSW/HPI capability since the SSF is only good for 72-hours. The extent of potential damage and single failures must be considered and addressed such that the capability to restore/use the PSW/HPI system is assured." "The licensing basis includes the capability to place the plant in cold shutdown and the mitigation strategy does not adequately address how this capability is assured relative to the extent of damage that can be experienced, recognizing that: a) it is critical to recover PSW/HPI within the 72-hour mission time of the SSF, and b) an assured source of cooling water that is good for at least 30-days is needed for the three Oconee units at the onset of tornado and HELB events."	As noted in our letter date 11/30/06, single active failures will be postulated, as appropriate, for initial event mitigation, to reach safe shutdown[1]. Should a single active failure occur in the PSW system during initial event mitigation, the SSF will be credited for initial event mitigation. Repair guidelines will be credited to restore the PSW system within the mission time of the SSF. Oconee agrees that the scope and detail of such repair guidelines needs to be described. These items will be provided in the LAR(s)	Agreed - Common Understanding - Additional information to be provided in the LAR. *[This item requests information to justify the statement that PSW system can be restored to operation within 72 hours. The HELB Report and the LAR reiterate this fact.]*
H18	**Seal Between the Reactor Building and the Auxiliary Building** "What is being done to assure that the seal between the reactor building (RB) and auxiliary building (AB) is properly maintained and does not leak excessively so that that flood mitigation features are not compromised?"	The seal between the RB and AB has been refurbished. This seal, as well as all other components required to prevent flooding of the AB following a MFDW break in the east penetration room, will be maintained as part of the station's civil passive features program (which is currently under development).	Agreed - Common understanding - No additional action

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H19	**Ability to Reach Cold Shutdown for Postulated HELBs** "...the licensing basis specifies the capability to place the Oconee units in cold shutdown condition and therefore, the licensee must be clear on what is being credited within the plant licensing basis in this regard such that the capability to achieve cold shutdown is assured." "The plant licensing basis is to be able to mitigate HELB events, including consideration of single active failures, and to place the plant in cold shutdown condition. The onus is on the licensee to demonstrate that the 72-hour mission time of the SSF is adequate for this purpose (e.g., extent of damage and time required to make necessary repairs and to resolve postulated failures of the PSW/HPI must be addressed)." "The plant licensing basis for both tornado and HELB includes the capability to achieve and maintain cold shutdown conditions. In the case of tornado, the station ASW system is credited for being able to maintain SSD for at least 30 days and the same capability should be provided by the PSW/HPI system. The submittal needs to explain how this capability will be assured, especially with respect to TS requirements." "The 72-hour mission time of the SSF does not establish what the mission time is for mitigating HELB scenarios. Adequate assurance must be established that the PSW/HPI and SSF are capable of mitigating the HELB event to the point of establishing cold shutdown conditions, irrespective of the SSF mission time. The 30-day capability of the PSW/HPI system can be credited in this regard, but assurance that sufficient water inventory will be available and that the PSW/HPI can be restored within 72-hours is required." "The plant licensing basis is to be able to mitigate HELB events, including the capability to place the plant in cold shutdown and consideration of single active failures. Loss of power is also postulated for those HELB events that can reasonably be expected to cause a loss of power, such as causing a trip of the main turbine." "The proposed licensing basis for HELB induced damage inside the TB indicates that no time-critical actions are required. The basis for this position is not obvious in that the SSF is only credited for 72-hours and the capability restore/use the PSW/HPI system prior to exceeding 72-hours is required. Also, the licensee needs to explain how a source of water for mitigating the HELB event is assured."	Adequate assurance will be provided in the unit specific LAR(s) that SSF or PSW/HPI can sustain the unit at safe shutdown[1] until cool-down can commence to cold shutdown. The LAR(s) will further demonstrate that an adequate source of water for SSF systems or PSW/HPI will be protected from HELBs and that the water inventory is adequate to sustain the function. It should be noted that for HELB events, crediting use of the submersible pump, the water can be supplied indefinitely (e.g. Lake Keowee). Should a single active failure occur on PSW/HPI, the SSF will be credited for initial event mitigation. Appropriate measures will be instituted and described in the unit specific LAR(s) to demonstrate that PSW/HPI can be restored within 72 hours. The equipment located inside the Turbine Building relied upon to establish cold shutdown is not protected from the effects of a HELB inside the Turbine Building. Subsequent to a HELB inside the Turbine Building, either the SSF or PSW/HPI system would be capable of providing secondary side decay heat removal and reactor coolant pump seal injection subsequent to a HELB event to maintain the affected units sub-cooled with a pressurizer steam bubble in safe shutdown[1] conditions for up to 72 hours. This mission time is consistent with the SSF current licensing basis. Additional damage repair may be required to enable the Low Pressure Service Water and the decay heat removal function of the Low Pressure Injection systems to achieve cold shutdown. For those events that cause loss of power, loss of power will be considered. There are no time critical operator actions inside the Turbine Building associated with plant cooldown or the establishment of cold shutdown.	Agreed - Common Understanding - Additional information to be provided in the LAR (Reference NRC Letter dated July 12, 2006, Enclosure 2, Item 2) *[Cold Shutdown is defined in Section 1.5 of the HELB Report (ONDS-351, R2) and the achievement of cold shutdown is identified as one of the shutdown intervals in Sections 3.5 and 3.6 of the HELB Report. The pathway for achieving cold shutdown for postulated HELBs in each unit is described in Sections 4.0, 5.0, and 6.0. Methodologies for achieving cold shutdown for specific postulated structural failures as a result of postulated HELBs are discussed in Section 8.0 of the HELB Report.]*
H20	**PSW/HPI Powering SSF** "HELB single active failure considerations rely to some extent upon the capability to align PSW/HPI power to the SSF. Therefore, contrary to the licensee's position as stated on Page 3 of Attachment 2, in Section 1.2, this capability should be included in the plant licensing basis." "HELB single active failure considerations rely upon the capability to align PSW/HPI power to the SSF. Therefore, contrary to the licensee's position (Section 1.2 on Page 3 of Attachment 2 of the submittal), it is necessary to credit this capability in the plant licensing basis."	The PSW/HPI power to the SSF is not necessary to mitigate a single failure within the initial 72 hours of the event. Therefore, it is Oconee's position that this function does need to be included in the licensing basis.	Agreed - Common Understanding - Additional information to be provided in the LAR *[The Duke response to this item said that crediting the PSW System to power the SSF is not necessary to mitigate postulated HELBs. Based upon the configuration of the PSW and SSF, the powering of the SSF with the PSW System for any postulated HELB is not necessary. This discussion is not in the HELB Report or any other part of the LAR.]*

ITEM #	NRC Issue	DUKE COMMENTS	RESOLUTION OF ITEMS
H21	**Main Steam Relief Valves (MSVRs) Cycling** "Page 3, first bullet: for how long and for how many cycles will the main steam relief valves (MSRVs) be credited; what assurance will be provided that they won't stick open, possibly compromising the mitigation strategy? What limitations are required relative to reactor coolant system (RCS) leakage when using the MSRVs and atmospheric dump valves (ADVs) for steam generator (SG) pressure control and crediting the SSF, and what changes are necessary to the TS in this regard?" "What are the maximum number of cycles the MSRVs will experience and why doesn't one or more MSRV sticking open pose a problem?" "The discussion indicates that steam pressure may be controlled using the ADVs to limit the number of MSRV cycles. What number of MSRV cycles are considered acceptable and why? What assurance is there that the MSRV cycles will be limited accordingly?"	Cycle test of a MSRV was completed 11/1/06. One thousand lift tests were conducted. At no time did the test relief valve stick open. Oconee views the results as a demonstration of the reliability of the valves to perform their design basis function during SSF operations. In addition, the number of lift tests conducted bounds the number of lift cycles expected during the 72 hour SSF mission time.	Agreed - Common Understanding - Additional information to be provided in the LAR *[This information is not in the HELB or Tornado LARs. Duke is not imposing limits on the MSRVs with regard to the maximum number of cycles. The CLB credits the cycling of the MSRVs for up to 72 hours.]*

Footnotes

{1} "Safe Shutdown" for the Oconee Nuclear Station is defined as Mode 3 with average Reactor Coolant System (RCS) temperature ≥ 525° F. "Cold Shutdown" is defined as Mode 5 with RCS temperature ≤ 200° F.

	NRC ISSUE	DUKE COMMENTS	Resolutions
T1	**USE OF TORMIS** a. "Page 2, second paragraph: Any differences in the design of Units 2 and 3 that could compromise the proposed tornado mitigation strategy that is based on Unit 1 design considerations need to be specifically identified and addressed." b. "The use of TORMIS must be requested in a LAR, and the TORMIS analysis should be applied to all SSCs that can adversely impact the tornado mitigation strategy, not just those SSCs that perform the functions that support the updated tornado mitigation strategy. For example, if a tornado missile ruptures an ammonia tank in the vicinity of the ADVs making it impossible to access the ADVs, then the ammonia tank would have to be included in the TORMIS analysis. Another example: if tornado missiles cause a structural failure of the TB that impacts the tornado mitigation strategy (such as by causing a failure of MS or other high energy piping), this would have to be included." c. "Pages 3/4, second bullet: The TORMIS analysis should be applied to all structures, systems, and components (SSCs) that can adversely impact the tornado mitigation strategy, not just those SSCs that perform the functions that support the updated tornado mitigation strategy. For example, if a tornado missile ruptures an ammonia tank in the vicinity of the ADVs making it impossible to access the ADVs, then the ammonia tank would have to be included in the TORMIS analysis. Another example: if tornado missiles cause a structural failure of the turbine building (TB) that results in a failure of main steam (MS) or other high energy piping that can compromise the tornado mitigation strategy, this would have to be included." d. "Second bullet: the use of TORMIS for must be requested in a LAR and the TORMIS analysis should be applied to all SSCs (safety and non-safety related) that can adversely impact the tornado mitigation strategy, not just those SSCs that perform the functions that support the updated tornado mitigation strategy (a complete listing of SSCs included in the TORMIS analysis is required). The NRC staff will allow the use of TORMIS provided it is consistent with what has been approved for use by other licensees. Any exceptions to the approved methodology, including modeling or analyses that are not included within the scope of TORMIS, will not be approved unless adequately justified." e. "Page 10, Section 5.2: *The TORMIS analysis must include all SSCs that can adversely impact the tornado mitigation strategy, not just those SSCs that perform the functions that support the updated tornado mitigation strategy, and "significant damage" would apply to all of these SSCs (e.g., damage to SSCs that can result in a main steam line failure and excessive cooldown; damage to SSCs that can prevent operators from taking required actions). *The proposed use of TORMIS must be requested and justified via an LAR; the previous approval does not apply to the current situation. *The TORMIS LAR will have to address anything that is beyond the scope of TORMIS approval, such as modeling considerations and damage assessment of specific SSCs (to the extent that this is utilized)." f. "Issue No. 1, "Use of TORMIS": *The proposed use of TORMIS must be requested and justified via an LAR; the previous approval does not apply to the current situation.* The TORMIS analysis must include all SSCs (safety-related and non-safety related) that can adversely impact the tornado mitigation strategy, not just those SSCs that perform the functions that support the updated tornado mitigation strategy; and "significant damage" would apply to all applicable SSCs in this regard (e.g., damage to SSCs that can result in a	a) All configurations described in the LAR will be validated for all three units prior to transmittal to the NRC. Additionally, the TORMIS analysis is being performed for all three units (although bounding arguments will be applied as possible) and any multi-unit interactions and interdependencies. b-g) Duke will describe how it intends to apply the TORMIS methodology in the LAR. Those components that are not or will not be protected from tornado missiles in accordance with UFSAR Class I or SSF missile criteria, will be evaluated with TORMIS. Attachment 2 of the Nov 30, 2006 letter describes the SSCs that are not designed to UFSAR missile criteria and the degree to which these SSCs are vulnerable. Attachment 2, Section 5.2 of the letter indicates that, in general, the analysis will collectively assess the ability of the SSF and PSW/HPI systems to meet the TORMIS acceptance criteria with respect to three functions 1) Secondary Side Decay Heat Removal 2) Reactor Coolant Pump Seal Injection and 3) Integrity of the Reactor Coolant System Pressure Boundary. The use of TORMIS was previously approved by the NRC for resolution of the secondary side decay heat removal GL-4 issue at ONS. The previously approved analysis is being extended to the reactor coolant pump seal injection and reactor coolant pressure boundary functions. There was no previous requirement to address the latter functions. However, TORMIS is being extended to these functions to add clarity and consistency to the LB. The analysis will be consistent with the five conditions (with the exception of the modified F-Scale) outlined in the SER's generic approval of the EPRI TORMIS methodology (dated Oct 28, 1983). An evaluation of secondary effects was not previously required for the resolution of the GL-4 issue or the IPEEE (see March 15, 2000 TER). Nonetheless, per Section 5.2 of Attachment 2 of the Nov 30, 2006 letter, ONS has committed to evaluating secondary effects in accordance with engineering judgment. Credit will be taken for activation of emergency response organizations and the assessment of plant conditions for any additional actions not specifically delineated in emergency operating procedures. As a general note, the Turbine Building contains approximately 4000 members in each building. As such, extensive damage by tornado missiles is not considered credible. h) TORMIS will be used to determine if any metal shielding will be added to protect SSF cabling leading into and through the CDTR and WPR. It will also be used to address an elevated trench on the north side of the SSF that is protected by a cantilevered section of the SSF facility. i) Initial TORMIS results indicate that the SSF will meet TORMIS acceptance criteria without reliance on PSW/HPI. Otherwise, these areas will be explicitly modeled by TORMIS since they support PSW/HPI operation.	a) Agreed - Common Understanding, additional information/detail to be provided in a LAR. The LAR will describe configurations for all 3 units. A list of SSCs (including mechanical, electrical and I&C components) that will be addressed by the TORMIS analysis will be included in the LAR. b-g) Agreed - Common understanding - additional information to be provided in the LAR. Duke will request the use of TORMIS in the LAR. The LAR will describe the application of the TORMIS methodology at ONS and include a list of tornado missile targets that will be evaluated for primary effects. Duke will address secondary effects using a qualitative assessment or TORMIS, as appropriate, in the LAR *[a-g) Provided as Attachment 4 of the Tornado LAR dated June 26, 2008.]* h) Agreed - Common understanding, no further action required. i) Agreed - Common understanding, no further action required.

	NRC ISSUE	DUKE COMMENTS	Resolutions
	main steam line failure and excessive cooldown; damage to SSCs that can prevent operators from taking required actions).* The TORMIS LAR will have to address anything that is beyond the scope of TORMIS approval, such as modeling considerations (including "secondary effects" modeling) and damage assessment of specific SSCs (to the extent that this is credited). g. "The TORMIS LAR will have to include a detailed listing of all SSCs that are included in the analysis, and address anything that is beyond the scope of the NRC staff's approval of TORMIS, such as modeling considerations and damage assessment of specific SSCs." h. "Commitments 3T and 4T: To what extent is TORMIS being used for this analysis?" i. "Page 9, Section 4: How will SSCs that are located in the cable spread, equipment, and control battery rooms be included within the scope of TORMIS? j. "The Oconee Updated Final Safety Analysis Report (UFSAR) states that the electrical equipment and cable rooms were constructed to UFSAR Class 1 structure tornado wind, differential pressure (DP), and missile criteria. This is a valid part of the plant licensing basis and it is consistent with the Oconee design criteria. The fact that these rooms were not constructed in accordance with the UFSAR description does not necessarily mean that the UFSAR is in error, but this may well be another licensing-basis discrepancy. Therefore, a change to the UFSAR in this regard must be properly evaluated and addressed in accordance with 10 CFR 50.59 requirements." k. Second bullet: the design details specified in the UFSAR that indicates that the electrical equipment and cable rooms were constructed to UFSAR Class 1 structure tornado wind, DP, and missile criteria is considered plant licensing basis and a change to the UFSAR in this regard must be addressed accordingly in accordance with 10 CFR 50.59 requirements."	j-k) In a September 15, 1986 letter, Duke stated that TORMIS analysis demonstrated that missile damage probability to all EFW and SSF ASW is less than the mean failure probability of 1E-6/rx-yr. The letter summarized the results of analyses assuming use of Station ASW. In the letter, Duke specifically noted that the Station ASW response time is 40 minutes, that the pressurizer safety valves (PSVs) will cycle to relieve pressure at 7 minutes and that the pressurizer will go water solid at 16 minutes. Additionally, the letter stated that "In light of the PRA result that the likelihood of EFW system failure due to tornado is very small, significant reliance on the Station ASW pump should not be considered necessary." Later, in a SER dated July 28, 1989, the NRC closed out the secondary side decay heat removal GL-4 issue. In the cover letter, the NRC stated that, "....the undamaged EFW system in one unit can supply feedwater to the steam generators in a unit with damaged EFW system cross-connections in the pump discharge piping." The cover letter concludes that, "Based on review of your probabilistic analysis, the staff concludes that the Oconee secondary side heat removal capability complies with the criterion for protection against tornadoes, and is therefore acceptable. This conclusion is <u>primarily based on the availability of the SSF ASW system</u>." For the purpose of tornado mitigation, the equipment and cable spread rooms support EFW and Station ASW. CLB depends on EFW from the unaffected unit but does not depend on Station ASW per the SER dated July 28, 1989 that resolved the secondary side decay heat removal GL-4 issue. The unaffected unit is not adversely impacted by the tornado. This will be addressed in accordance with 50.59 requirements.	j-k) Agreed - Common Understanding, additional information/detail to be provided in a LAR. *[j-k) Provided as Attachment 4 of the Tornado LAR dated June 26, 2008]*
	l. Page 9, Section 4: How will tornado missile capability to fail TB operating deck be addressed by the analysis?" m. "Page 3, first paragraph: the use of physical separation or physical barriers to protect one or more of the systems is not entirely accurate in that a TORMIS analysis will also be used." n. "Page 5, Section 1.5: How will the TORMIS analysis evaluate turbine building structural failures that are sufficient to cause MS pipe or other high energy pipe failures, thereby compromising the tornado mitigation strategy?"	l) Given the construction and configuration of the Turbine Building operating deck, failure of the deck due to missiles is not considered credible. m) The discussion related to physical separation is included in the Nov 30 2006 letter to demonstrate why the addition of the PSW/HPI system reduces risk relative to tornado missile damage in a subjective manner. n) See Item l	l, n) Agreed - Common Understanding, additional information/detail to be provided in a LAR. The LAR will include an evaluation for the Turbine Building Structure and Operating Deck for damage due to tornado missiles that could significantly impact the tornado mitigation strategy. *[l, n) The aforementioned information was captured in the revised UFSAR pages shown in Attachment 3 of the Tornado LAR.]* m) Agreed - Common Understanding, no further action required.
	o. "Page 8, Section 3: what part of the CCW piping is not protected from tornado missiles, and is it being evaluated by TORMIS?" p. "Vulnerable CCW piping should be included in the TORMIS analysis." q. "Station modifications that provide reinforcement of an expansive portion of key structures to better withstand the effects of tornados - use of fiber reinforced polymer. What structures will be protected?" r. "Commitment 5T: How will a tornado missile strike that compromises the fiber reinforced polymer be addressed in the TORMIS analysis?" s. Page 5, Section 1.6: Is existing plant vital I&C power tornado protected; and are power sources for PSW/HPI vulnerable to tornado effects?"	o-p) A limited amount of CCW piping in the basement of the Turbine Building is not protected from tornado missiles. An evaluation will be performed to demonstrate that failure of this piping is not credible. q) The WPR and CDTR walls will be upgraded via FRP. r) The FRP is being added as an enhancement for tornado wind and DP. It is not being credited for missile protection.	o-p) Agreed - Common Understanding, additional information/detail to be provided in a LAR. *[o-p) Additional information is provided in Duke Energy's Tornado LAR and subsequent RAI responses dated 5/6/10, 5/25/10, 6/24/10, and 8/31/10.]* q) Agreed - Common Understanding, no further action required. r) Agreed - Common Understanding, additional information/detail to be provided in a LAR. *[r) TORMIS modeling assumptions related to the West Pen Room walls are described in Section 5.4 in Attachment 4 of the Tornado LAR dated June 26, 2008.]*

	NRC ISSUE	DUKE COMMENTS	Resolutions
		s) The new switchgear for PSW/HPI will be enclosed in a tornado protected enclosure. There is a limited vulnerability to tornado missiles in the equipment, control battery and cable spread rooms. The rooms are largely protected by adjacent structures. The SSF vital I&C are fully protected in the SSF facility and provide redundancy to the PSW/HPI system. In general, PSW/HPI instrumentation enters containment through the EPR and SSF enters containment through the WPR. Exceptions relate to PSW to the SG through the WPR (however, this only provides backup to the other PSW train in the EPR and SSF ASW in the WPR).	s) Agreed - Common Understanding, no further action required.
T2	**COLD SHUTDOWN** a. The PSW/HPI capability is the only means that can be relied upon for tornado and HELB mitigation beyond 72-hours, and it is the only means available for cooling down the Oconee units." b. "Fifth bullet: the licensing basis specifies the capability to place the Oconee units in cold shutdown condition and therefore, the licensee must be clear on what is being credited within the plant licensing basis in this regard such that the capability to achieve cold shutdown is assured." c. "Issue No. 2, "Cold Shutdown" * The plant licensing basis for both tornado and HELB includes the capability to achieve and maintain cold shutdown conditions. In the case of tornado, the station ASW system is credited for being able to maintain SSD for at least 30 days and the same capability should be provided by the PSW/HPI system. The submittal needs to explain how this capability will be assured, especially with respect to TS requirements. d. "Issue No. 2, "Cold Shutdown":** The licensing basis includes the capability to place the plant in cold shutdown and the mitigation strategy does not adequately address how this capability is assured relative to the extent of damage that can be experienced, recognizing that: a) it is critical to recover PSW/HPI within the 72-hour mission time of the SSF, and b) an assured source of cooling water that is good for at least 30-days is needed for the three Oconee units at the onset of tornado and HELB events." e. "Page 2, fourth paragraph: The manual alignment of the spent fuel pool (SFP) to HPI is a change to the original licensing basis that was not submitted for NRC review and approval." f. "Third bullet: the spent fuel pool (SFP) to the HPI pump flow path that was established by Duke after initial licensing of the Oconee units was not	a-d) ONS can find no evidence within the UFSAR or licensing correspondence with the NRC that would indicate that ONS has committed to achieve cold shutdown within specific time for tornado mitigation. Although the UFSAR does indicate that ONS has over 30 days of secondary heat removal inventory, it does not indicate that the SSF or other systems are capable of sustaining secondary heat removal without reliance on additional actions. The SSF mission time, for instance, is 72 hours in accordance with the SSF SER date April 28, 1983 and the GL-4 issue SER dated July 28, 1989. As indicated in Attachment 1, Commitment 7T, 5th bullet, ONS will enhance existing damage repair guidelines to extend the 72 hour safe shutdown capability of the SSF and to establish cold shutdown conditions. This enhanced capability will not be part of the LB. e-f) The SFP-HPI flow path will be removed by the LAR.	a-d) Specific aspects of the damage repair guidelines to extend the 72 hour safe shutdown capability of the SSF and to establish cold shutdown conditions will be described in the LAR. Agreed - Common understanding, additional information/detail to be provided in LAR. *[a-d) The aforementioned information is described in Enclosure 2, Section 4.2, "Damage Repair Guidelines and Procedures," of the Tornado LAR. Additional information is provided in T8 and Duke Energy's Tornado LAR RAI responses dated 5/6/10, 5/25/10, 6/24/10, and 8/31/10.]* e-f) Agreed - Common understanding, no further action required.

	NRC ISSUE	DUKE COMMENTS	Resolutions
	submitted for NRC review and approval."		
T3	**TECHNICAL SPECIFICATIONS** a. TS requirements that assure the operability and availability of structures, systems and components (SSCs) that are relied upon for the tornado and HELB mitigation strategies must be established, such as for the standby shutdown facility (SSF), the PSW/HPI system, the Unit 2 condenser circulating water (CCW) system, and for reactor coolant system leakage. b. "No mention of establishing a Technical Specification (TS) requirement pursuant to 10 CFR 50.36 even though the licensee recognizes that the PSW/HPI modifications will "significantly" improve overall plant risk." c. "Tornado and HELB events at Oconee represent at least the same level of risk as associated with design basis accidents (DBAs)." d. "The licensee proposes to rely upon the PSW/HPI system in conjunction with the SSF for tornado and HELB mitigation, and the licensee's TORMIS analysis is predicated on this. Therefore, TS requirements should be established not only to assure the operability of the PSW/HPI system, but also to assure that both the SSF and PSW/HPI systems are not both rendered inoperable at the same time." e. "The existing TS AOT must be reconsidered accordingly recognizing the new tornado and HELB mitigation functions." f. Operability of the water inventory for the SSF and PSW/HPI must be addressed. The current SSF TS in this regard was based on the availability of other systems such as EFW for performing the SSDHR function, which is not valid for the proposed tornado and TB HELB mitigation strategies. Furthermore, both the SSF and the PSW/HPI systems rely upon the same water supply and the licensee has not addressed how the water supply will be assured for both tornado and HELB mitigation." g. "First bullet: licensing basis clarity should be reflected in the Updated Final Safety Analysis Report (UFSAR), and TS requirements should be established in accordance with 10 CFR 50.36 requirements." h. "Issue No. 3, "Technical Specifications" A TS is required for the PSW/HPI system in accordance with 10 CFR 50.36(c)(2)(ii)(D). As stated on Page 3 of the November 30, 2006, submittal, Duke indicated that "the installation of a new PSW system and HPI system improvements will reduce reliance on the SSF by providing a system capable of independently establishing safe shutdown conditions, and thereby significantly improve overall plant risk." TS requirements were required for the SSF to assure its SSDHR function (even though other sources of SSDHR were considered to be available). The risk significance of the PSW/HPI system is on the same order of magnitude as the SSF and in this case, other sources of SSDHR may not be available." i. "Page 3, first bullet: Relative to the capability to power the submersible pump	a-l) See HELB, Issue H10	a-l) See HELB, Issue H10

	NRC ISSUE	DUKE COMMENTS	Resolutions
	by the PSW switchgear, what TS operability and surveillance requirements are appropriate?" j. "How is capability of submersible pump (powered by either SSF or PSW/HPI) assured by TS requirements?" k. "Page 3, first bullet: SSF and PSW will both use the Unit 2 condenser circulating water (CCW) inlet piping as a water source. How will availability of this water source be assured? The current SSF TS requirements did not include consideration of the proposed mitigation strategy and the current 45 day allowed outage time (AOT) for the Unit 2 CCW inlet is of concern. This needs to be reconsidered since the basis for the 45 day AOT is no longer valid, and the AOT should be limited based on tornado and HELB considerations recognizing that there are not other sources of water." l. How is water supply from the Unit 2 CCW assured to be available?		
T4	<u>REACTOR COOLANT LETDOWN LINE</u> a. "It is the NRC staff's position that the plant licensing basis for postulated failures of the letdown line includes consideration of single active failures, and postulated failures of the letdown lines for the Oconee units must be addressed accordingly."	a) Section 5.2 of Attachment 2 of the Nov 30, 2006 letter indicates that TORMIS will be used to evaluate the integrity of the reactor coolant system pressure boundary.	a) Agreed - Common Understanding, no further action required.
T5	<u>OPERATOR ACTIONS</u> a. "In order for the SSF to be credited, operators would have to be dispatched to the SSF during a tornado watch, not during a tornado warning as proposed. Once a tornado watch has been declared, the only question that remains is whether or not the tornado will touch down at Oconee or someplace else. If this is the one that hits Oconee, the SSF would not be accessible and it would be too late at this point to man the SSF until the tornado has passed." b. "Page 3, Section 1.2: Operators should be dispatched to the SSF during a tornado watch. A tornado warning means that the tornado has already touched down and it would be too late at this point to man the SSF if this turns out to be the tornado that hits the Oconee site. c. "Page 13, Section 7: The SSF should be manned upon declaration of a tornado watch. A tornado warning means that the tornado has already touched down and it would be too late at this point to man the SSF if this turns out to be the tornado that hits the Oconee site." d. "Why aren't PSW/HPI design criteria and time critical actions included similar to HELB commitments that were made?"	a-c) Response provided at Region II Pre-Decisional Conference Related to Unit 3 Control Room North Wall. Duke developed an event tree analysis to evaluate affects of tornado warning time. The ONS natural disaster procedure dispatches operators to the SSF upon receipt of tornado warning notification. The average response time is 3.6 minutes and the average travel time to SSF is 4 minutes. Based on National Weather Service (NWS) data, average tornado warning time is 13 minutes. Oconee believes there is minimal impact on overall SSF reliability. Note: Tornado warnings include identification via Doppler Radar. d) No comment.	a-c) Agreed - Common understanding, additional information/detail to be provided in LAR. The average warning time subsequent to issuance of a tornado warning and the average operator response time required to man the SSF subsequent to a tornado warning will be described in the LAR. d) Agreed - Common understanding, additional information/detail to be provided in LAR. *[a-d) Information on operator actions is provided in Duke Energy's tornado LAR (enclosure 2) and subsequent tornado RAI response dated 6/24/10.]*
T6	<u>MSRV CYCLING</u> a. "Page 3, Section 1.2: The discussion indicates that steam pressure may be controlled using the ADVs to limit the number of MSRV cycles. What number	a-c) See response to questions under HELB Issue H21.	See HELB Issue H21.

	NRC ISSUE	DUKE COMMENTS	Resolutions
	of MSRV cycles are considered acceptable and why? What assurance is there that the MSRV cycles will be limited accordingly?" b. "What limitations are required relative to reactor coolant system (RCS) leakage when using the MSRVs and atmospheric dump valves (ADVs) for steam generator (SG) pressure control and crediting the SSF, and what changes are necessary to the TS in this regard?" c. "Page 2, Section 1.1: What are the maximum number of cycles the MSRVs will experience and why doesn't one or more MSRV sticking open pose a problem?		
T7	**PSW DESIGN ISSUES** a. "Page 3, Section 1.3: what impact does tornado missile damage to the PSW piping in one penetration room have on the capability of PSW/HPI to perform its functions?" b. "Page 6, Section 2.3: what impact does damage to piping/electrical/I&C in one penetration room have on tornado mitigation capability of PSW/HPI? What is the effect on other units? Similarly for SSF?" c. "Page 8, Section 2.5: is any of the PSW I&C power not tornado protected?" d. "Page 9, Section 5.1: In addition to protecting the SSF and PSW/HPI components "that perform the functions," what about any support equipment that is needed (I&C, ADVs, RCP SI, etc.)?" e. "The installation of a new protected service water (PSW) system with switchgear capable of providing an assured source of electrical power to (among other things) the high pressure injection (HPI) pumps. Contrary to the information that was provided, this PSW/HPI system is not totally independent of the standby shutdown facility because they share the same water source." f. "Indicates that installation of PSW and HPI improvements will reduce reliance on the SSF by providing a system capable of independently establishing safe shutdown conditions, thereby significantly improving overall plant risk - not truly independent due to shared water source and west penetration room (WPR) vulnerabilities." g. "Sixth and seventh bullets: the commitment should specify that the PSW/HPI and related switchgear modifications will satisfy safety-related, seismic Category 1 criteria, and will be controlled and maintained in accordance with 10 CFR 50, Appendix B criteria." h. "Clarifications Required Concerning the Tornado and HELB Mitigation Strategies: "While the licensee seems to suggest that the PSW/HPI system will be installed as safety-related, seismic Category 1, and will be controlled in accordance with 10 CFR 50 Appendix B requirements, this needs to be clearly stated to assure that there is no misunderstanding."	a) The PSW supply to each SG is physically separated by containment. Either supply is adequate for secondary heat removal. SSF ASW also provides defense-in-depth. b) Preliminary TORMIS analysis indicates that SSF meets TORMIS criteria without reliance on PSW/HPI. As such, PSW/HPI provides margin to uncertainties. The description of physical separation provides additional qualitative assurance of the added value of PSW/HPI. c-d) See Item s under Issue T1. e-f) See HELB Issue, H21. g-h) See Common Issue, C1.	a-b) Agreed - Common Understanding, no further action required. c-d) See Item s under Issue T1. e-f) See HELB Issue, H21. g-h) See Common Issue, C1.

	NRC ISSUE	DUKE COMMENTS	Resolutions
T8	**CONCURRENT DAMAGE TO KHU/STATION SWITCHYARD** a. "Clarifications Required Concerning the Tornado Mitigation Strategy: *In addition to the specific tornado effects that the licensee referred to, the following additional considerations are also applicable: a complete loss of offsite power; and while the tornado is not assumed to cause tornado missile damage to the Keowee Hydro Units (KHU) and the Oconee units concurrently, it is assumed that both KHU and the Oconee units can be exposed to tornado force winds concurrently." b. In addition to the tornado effects that the licensee referred to, the following additional considerations also apply: the tornado effects include a complete loss of offsite power, and while the tornado is not assumed to cause tornado missile damage to KHU and the Oconee units concurrently, it is assumed that KHU is exposed to the tornado-force winds that would exist; and vice-versa for a tornado striking KHU. c. "Page 12, Section 6: The plant licensing basis includes the capability to achieve cold shutdown. The EDGs for other plants provide a 7-day capability to restore offsite power or to establish additional fuel oil inventory. The proposed 72-hour capability is not commensurate with the 7-day capability that is provided by other plants and the extensive damage that can be caused to the electrical distribution network in the vicinity of the Oconee station following a tornado strike at KHU could require well beyond 72-hours to restore a normal source of electrical power. Therefore, in order to assure the capability to maintain safe shutdown conditions and to subsequently achieve cold shutdown, the PSW/HPI mods should also include consideration of a tornado-protected capability to connect a temporary power source within 72-hours that is adequate for powering the PSW/HPI functions. Also note that there is no mention of how SFP makeup and boron addition will be accomplished over an extended period of time." d. "Page 2, first paragraph: In addition to the tornado effects that the licensee referred to, the following additional considerations are also applicable: a complete loss of offsite power; and while the tornado is not assumed to cause tornado missile damage to the Keowee Hydro Units (KHU) and the Oconee units concurrently, it is assumed that KHU is exposed to the tornado-force winds and vice-versa for a tornado striking KHU." e. Page 5, Section 1.6: * The capability to install (via a tornado protected connection) and use temporary power within 72 hours should also be considered since PSW/HPI is relied upon exclusively for maintaining SSD beyond 72-hours and for plant cooldown. f. Page 5, Third Bullet: This is taken out of context; the SSF auxiliary service water (ASW) system was specifically credited for mitigating the tornado that damages KHU with concurrent LOOP. Otherwise, the NRC SE accepted the licensee's analysis that credited station ASW and emergency feedwater (EFW) from the unaffected units." g. "Page 2, third paragraph: The Oconee current licensing basis (CLB) does not rely "extensively" on the SSF. This is only the case for when the tornado strikes KHU resulting in a loss of power to the Oconee station. Otherwise, Station ASW and EFW of the other unaffected units was relied upon in the CLB."	a-e) The original and current UFSAR refers to physically separated power supplies that include KHU and the station switch yard. As an enhancement, an alternate power supply is being installed from the Lee CT 100 KV line to the PSW protected switchgear to further reduce the probability of a loss of power to the PSW/HPI system in the event of a coincident strike of the Station and Keowee. The probability of coincident tornado damage to the Station and Keowee was previously assessed in the ONS IPEEE. See commitment 7T last bullet Attachment 1 and Sections 1.6 and 2.5 of Nov 30, 2006 letter. Cold shutdown aspects discussed under Issue T2. Spent fuel pool makeup is currently addressed by SSF operational procedures. f-h) See Items j-k under Issue T1 Note: In conclusion, from a licensing perspective, the PSW system will replace the EFW system from the unaffected unit. In addition, the tornado event will be conservatively considered a 3 unit versus a single unit event.	a-e) Agreed - Common understanding, additional information/detail to be provided in LAR concerning the Lee 100 kV line and the zone of influence of the tornado path. The LAR will include information regarding the Lee CT 100 KV line. See July 12, 2006 NRC letter, Enclosure 2, Item 5. *[a-d) Additional information is also provided in Duke Energy's Tornado LAR (enclosure 2, Revised licensing basis - emergency power) and subsequent RAI responses dated 5/6/10, 5/25/10, 6/24/10, and 8/31/10.]* *[e) The use of temporary electrical power beyond the SSF's credited 72 hours was considered but not proposed in the overall mitigation response submitted to the Staff in the Tornado LAR.]* f-h) See Items j-k under Issue T1.

	NRC ISSUE	DUKE COMMENTS	Resolutions
	h. *"Issue No. 2, "Cold Shutdown" : The CLB relies upon SSF for providing secondary side decay heat removal (SSDHR) only when the tornado takes out KHU; otherwise station ASW is relied upon for long-term cooling."		

Response Documentation for:

Item 62

3.7 PLANT SYSTEMS

3.7.10 Protected Service Water (PSW) System

LCO 3.7.10 The PSW System shall be OPERABLE as follows:

a. The mechanical portion of the PSW System is OPERABLE,
b. *The electrical portion of the PSW System is OPERABLE including a* power supply to the PSW switchgear from either:
   1) The KHU Protected Service Water Power Path or,
   2) The 100 kV Central Tie Switchyard overhead line.

Deleted: KHU underground path

APPLICABILITY: MODES 1, 2, and 3
MODE 4 when steam generators are relied upon for heat removal.

ACTIONS

-----NOTE-----
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PSW System is INOPERABLE. AND SSF Systems are OPERABLE.	A.1 Restore PSW System to OPERABLE status.	30 days
B. PSW System is INOPERABLE. AND SSF Systems are INOPERABLE.	B.1 Restore PSW System to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met when the PSW System is INOPERABLE due to maintenance.	C.1 Restore to OPERABLE status.	-----------NOTE---------- Not to exceed 45 days cumulative per calendar year ------------------------------ 45 days from discovery of initial inoperability.
D. Required Action and associated Completion Time of Condition C not met. OR Required Action and associated Completion Time of Condition A or B not met for reasons other than Condition C.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify required PSW battery terminal voltage is ≥ 125 VDC on float charge.	7 days
SR 3.7.10.2	Verify that the KHU Protected Service Water Path can be aligned to and power the PSW electrical system.	90 days
SR 3.7.10.3	Verify that the developed head of the PSW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the In-Service Testing (IST) Program
SR 3.7.10.4	Verify for required PSW battery that the cells, cell plates and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	12 months

Deleted: KHU underground

(continued)

BASES (continued)

LCO (continued)

- RCS and Reactor Vessel Head high point vent valves
- PSW electrical system from either the KHU Protected Service Water Path or 13.8 kV overhead power paths to support secondary side decay heat removal (SSDHR) and reactor coolant make-up (RCMU) functions.
- A required number of 125 VDC Vital I&C Normal Battery Chargers (Ref.: TS 3.8.3, DC Sources – Operating).

Deleted: KHU underground

PSW system dedicated instrumentation and controls located in each main control room:

- Two (2) high flow controllers (one per SG).
- Two (2) low flow controllers (one per SG).
- One (1) flow indicator (per SG).
- Two (2) SG header isolation valves (one per SG header).
- Two (2) HPI System power transfer switches per unit.
- Power transfer switches to HPI valves needed to align the BWST to the HPI pumps.

APPLICABILITY

In MODES 1, 2, and 3, the PSW System is required to be OPERABLE and to function in the event that all normal and emergency feedwater systems are lost. In MODE 4, with RCS temperature above 212 °F, the PSW System may be used for heat removal via the steam generators. In MODE 4, the steam generators are used for heat removal unless this function is being performed by the Low Pressure Injection System. In MODE 4 steam generators are relied upon for heat removal whenever an RCS loop is required to be OPERABLE or operating to satisfy LCO 3.4.6, "RCS Loops – Mode 4."

In MODES 5 and 6, the steam generators are not used for SSDHR and the PSW System is not required.

ACTIONS

The exception for LCO 3.0.4, provided in the Note of the Actions, permits entry into MODES 1, 2, 3 or 4 with the PSW not OPERABLE. This is acceptable because the PSW is not required to support normal operation of the facility or to mitigate a design basis accident.

source for the HPI pump or alternatively, the pump can be manually aligned to a SFP should the BWST be unavailable. For the SSF RCMU pump, water from the SFPs is used and RCS inventory is managed from the SSF CR.

As described in UFSAR Sections 3.3.2 and 3.8.4.3, certain structures that house systems and components necessary to achieve SSD have been constructed to withstand the effects of a tornado (wind, ΔP, and missiles). Other specific structures necessary to achieve SSD, while designed to withstand wind and ΔP, were evaluated for the probability of a damaging missile strike using risk analysis. An example of the latter includes the WPR walls. Longer-term recovery actions beyond the current SSF 72 hour mission time are not addressed in the CLB.

Revised LB

The overall objective of the revised tornado LB is to utilize the SSF for SSDHR and RCMU following a loss of all normal and emergency systems which usually provide these functions. The SSF systems can maintain all three units in a safe shutdown condition, i.e., Mode 3 with average RCS temperature ≥ 525 °F (unless the initiating event causes the unit(s) to be driven to a lower temperature[15]) for up to 72 hours while damage control measures are completed to restore any unavailable PSW System equipment needed to cooldown the units to ~250 °F. This mission time is in accordance with the SSF CLB. The ~250 °F temperature is the lowest that can be attained using the steam generators (SGs) for cooldown.

The existing Station ASW system will be replaced with a new PSW system and be capable of cooling the units to approximately 250 °F where they would remain until additional damage control measures can facilitate cooldown to cold shutdown (CSD)[16] conditions. Although the SSF or the new PSW systems both have the capability to restore SSDHR and RCMU for all three units, the PSW system is not fully protected from a severe tornado and as such, is not credited in the revised LB within the first 72 hours after a tornado.

The revised tornado LB assumes that a tornado strikes the plant site during full power operation and disables the emergency and non-emergency electrical buses located in the TB resulting in a station blackout condition. A further assumption is that due to the approximate ¾ mile separation between the KHUs and the Oconee Nuclear Units, a tornado missile will not cause concurrent damage to both the KHUs and the Oconee Nuclear Units. As added margin, alternate power (primary power is from the KHU Protected Service Water Path) to the new PSW System is provided from the Central Tie

Deleted: KHU underground feed

[15] TORMIS results (Attachment 4) have shown that the probability of a damaging missile striking the MS lines upstream of the new MSIVs to be extremely low and as such, there is reasonable assurance that a rapid RCS cooldown transient resulting in RCS temperatures falling below the SSF threshold temperature, to be remote. Therefore, tornado induced MS line breaks are not postulated in the revised tornado mitigation strategy.

[16] Cold shutdown is Mode 5 with RCS temperature < 200 °F.

Emergency Power

Current LB

A protected diesel generator supplies power to the SSF and its support systems for up to 72 hours. The SSF power supply system is designed to provide normal and independent emergency sources of AC and DC electrical power to their associated electrical distribution systems and various support systems. The SSF diesel generator would only be operated in the event where normal power systems are unavailable. Manual operator action is required to actuate the SSF.

Power to the Station ASW switchgear, located below grade in the AB, is supplied from the KHU Protected Service Water Path. This switchgear can power a Station ASW pump and one HPI pump per unit. The structures that comprise the KHUs are the Powerhouse, Power and Penstock Tunnels, Spillway, Service Bay Substructure, Breaker Vault, and Intake Structure. The KHUs are Class 2 structures which have not been designed and built to resist tornado loads. At ONS, the wind loading of a Class 2 structure is 95 miles per hour.

Deleted: KHU underground feed

Revised LB

A protected diesel generator supplies power to the SSF and its support systems for up to 72 hours. The SSF power supply system is designed to provide normal and independent emergency sources of AC and DC electrical power to their associated electrical distribution systems and various support systems. The SSF diesel generator would only be operated in the event where normal power systems are unavailable. Manual operator action is required to actuate the SSF systems.

The Station ASW switchgear will be replaced with the PSW System switchgear located in a new tornado-protected PSW building. New power cables will be routed from the KHUs to the PSW building through an underground path. Alternate power to the PSW System switchgear will be provided by a new transformer connected to the existing 100 kV transmission line that receives power from the Central Tie Switchyard located approximately 8 miles from the plant. This new power path is strategically located on the opposite side of the station from the KHUs which reduces the chance of concurrent tornado damage to both power sources.

The new tap-off portion from the 100 kV line will not adversely affect the operation of the station's CT5 emergency transformer. Any fault that occurs on this new portion of line will be isolated from the 100 kV line with either the high side circuit switcher or the low side breaker installed at the PSW substation. The PSW switchgear will also provide a backup power supply to the SSF via an underground path as additional defense-in-depth. An electrical diagram displaying the revised power arrangement for the SSF and PSW Systems and the location of the CT5 transformer is shown on Figure 1.

Although the power lines from the alternate offsite power supply to the PSW switchgear

4.5 Conclusions

Implementation of the revised tornado LB and the related commitments will clarify and, in some cases, revise the ONS CLB to address issues raised by the NRC and collectively enhance the station's overall design, safety and risk margin. The safety margins afforded by the revised tornado LB will be improved by:

- Verification that the SSF is the assured means of achieving SSD conditions for one, two, or all three units.
- Replacing the single-unit low-head Station ASW system with a 3-unit high-head PSW System that:
  - is controllable from the main CRs,
  - can be placed into service quickly to minimize inventory loss from the PZR safety valves,
  - increases assurance that natural circulation will be established and maintained,
  - can be powered from either the KHU Protected Service Water Path or alternatively, the 100 kV Central substation path located on the opposite side of the station from the KHUs which reduces the chance of concurrent tornado damage to both emergency power sources.
- Physically protecting the BWST to the extent necessary, to assure that the tank and flowpath are available following a tornado.
- Installation of MSIVs for each unit's main steam header.
- The elimination of several time-critical manual operator actions outside of the CRs including:
  - ADV operation for SG depressurization,
  - Alignment of the Station ASW valves and breakers,
  - Connection of the Station ASW switchgear power supply to an HPI pump and,
  - Alignment of the SFP to HPI flow path.

Deleted: KHU underground