Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

June 22, 2011

L-2011-232
10 CFR 50.90

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

Re: St. Lucie Plant Unit 1
Docket No. 50-335
Renewed Facility Operating License No. DPR-67

Response to NRC Balance-of-Plant Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2010-259), "License Amendment Request for Extended Power Uprate," November 22, 2010, Accession No. ML103560419.
(2) Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie Unit 1 EPU – request for additional information (Balance of Plant)," May 27, 2011, Accession No. ML111510227.

By letter L-2010-259 dated November 22, 2010 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. DPR-67 and revise the St. Lucie Unit 1 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an Extended Power Uprate (EPU).

By email from the NRC Project Manager dated May 27, 2011 [Reference 2], additional information related to the balance-of-plant was requested by the NRC staff in the Balance of Plant Branch (SBPB) to support their review of the EPU LAR. The request for additional information (RAI) identified eight questions. The response to these RAIs is provided in Attachment 1 to this letter.

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NRR

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-259 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-429-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on 22-June-2011

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment

cc: Mr. William Passetti, Florida Department of Health

Response to Request for Additional Information

The following information is provided by Florida Power & Light (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Nuclear Plant Unit 1 that was submitted to the NRC by FPL via letter (L-2010-259) dated November 22, 2010, Accession Number ML103560419.

In an email dated May 27, 2011 from NRC (Tracy Orf) to FPL (Chris Wasik), Accession Number ML111510227, Subject: St. Lucie Unit 1 EPU – request for additional information (Balance of Plant), the NRC requested additional information regarding FPL's request to implement the EPU. The RAI consisted of eight (8) questions from the NRC's Balance of Plant Branch (SBPB). These eight RAI questions and the FPL responses are documented below.

SBPB-1: 2.5.1.1.1-01, Flood Protection

The NRC's acceptance criteria for flood protection are based on General Design Criteria (GDC) -2, which states that "Structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability to perform their safety functions." The licensee stated in its licensee amendment request (LAR) that GDC-2 addresses only external flooding analysis. The licensee referenced a submittal of an Individual Plant Examination (IPE) for St. Lucie, Units 1 and 2, to the staff in 1992 in response to the Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities – 10 CFR 50.54 (f)." The IPE submittal addressed the internal flooding analysis for St. Lucie, Units 1 and 2.

In the LAR, the licensee stated that the internal flooding analysis in the St. Lucie IPE identified "flood zones" for safety-related areas and/or equipment, which could potentially contribute to the overall core damage frequency in the event of flooding. In particular, areas such as auxiliary feedwater pumps and switchgear rooms were identified as part of the internal flooding analysis. Although the licensee stated that the extended power uprate (EPU) would not affect the current internal flooding analysis for the safety-related areas as provided in the IPE, the licensee did not provide justification to address how each of the "flood zones" were assessed according to the IPE to make its determination.

Provide additional justification regarding internal flooding analysis for the "flood zones" under EPU conditions.

Response to SBPB-1

As addressed in the IPE submittal, for screening purposes, a bounding flood and/or spray scenario was postulated within a zone. If it was concluded that the resulting flood level or spray could cause failure of a component(s), the effect of loss of the specific component(s) was then analyzed in terms of whether the affected component is associated with an initiating event (an event causing a demand for a reactor trip) and/or equipment relied upon to mitigate the consequences of an accident (probabilistic risk assessment [PRA] equipment). A flood zone was initially "screened out" if flood or spray would not cause an initiating event and damage PRA equipment at the same time. For the flood zones not screened out based on this initial screening, screening based on a qualitative and/or quantitative assessment was performed.

For the flood zones not screened out based on a qualitative and/or quantitative assessment, calculation / quantification of the contribution to core damage frequency (CDF) is discussed in the IPE submittal.

As addressed in the IPE submittal, one notable factor in several assessments is the role of plant operators in flood detection. These individuals travel pre-determined circuits at a prescribed frequency to take log readings on numerous plant components. Given their training and responsibility for vigilance, it is likely that they would notice flood waters or evidence of flooding during their rounds. Thus, several of the qualitative assessments take credit for early detection of flooding by plant operators.

The following describes the qualitative / quantitative assessment of the applicable flood zones in the 1993 IPE submittal and evaluates the impact of the extended power uprate (EPU) on the discussion / results of the IPE assessment of each zone:

<u>Steam Trestle / Auxiliary Feedwater (AFW) Pumps Area</u>

<u>IPE Assessment</u>

On one side of the steam trestle area, the feedwater line is directly over both motor-driven AFW pumps. On the other side of the steam trestle area, the other feedwater line is directly over the turbine-driven AFW pump. Rupture of a feedwater line could fail either (a) both motor-driven AFW pumps, or (b) the turbine-driven AFW pump. These scenarios could not be screened out qualitatively, and therefore calculation of the contribution to CDF is discussed in the IPE submittal.

<u>EPU Impact</u>

The EPU does not affect the locations of the AFW pumps or the locations of the feedwater lines in the AFW pumps area. The EPU does not affect the conclusion that rupture of a feedwater line could fail either (a) both motor-driven AFW pumps, or (b) the turbine-driven AFW pump due to spray damage. Therefore, the EPU does not affect the IPE assessment of the steam trestle / auxiliary feed pumps area.

<u>Intake Cooling Water (ICW) Area</u>

<u>IPE Assessment</u>

The concern in this area is if one ICW pump discharge line sprays the adjacent pump motor, and the third pump is out of service, all ICW could be lost. An event of this nature is not deemed credible based on the following: (a) this event could only happen when the A or B pump are out of service, since C is the middle pump; since the A and B pumps are the normally running pumps, operation with the A or B pump out of service occurs only during a small fraction of the time for power operation, (b) only a small class of pipe breaks would have the right size, geometry, and location to spray an adjacent pump's motor, and (c) due to the motor's construction, it is unlikely that spray would contact electrical parts.

EPU Impact

The EPU does not affect the locations of the ICW pumps or the construction of the ICW pump motors. Currently, ICW pumps A, B, and C are operated such that pump run times are equalized; the EPU does not affect operation of the ICW pumps. The EPU does not affect the ICW pump head performance. The EPU does not affect the conclusion that only a small class of pipe breaks would have the right size, geometry, and location to spray an adjacent pump's motor. Therefore, the EPU does not affect the IPE assessment of the ICW area.

Component Cooling Water (CCW) Area

IPE Assessment

The CCW pit contains ICW and CCW piping. Several barriers to the complete flooding of this area exist.

Low flow breaks, or "cracks," have been analyzed in the Unit 2 FSAR, Appendix 3.6F; the analysis is assumed to be essentially equivalent for Unit 1. The analysis in the Unit 2 FSAR evaluates flooding from a crack in the ICW system inlet line to the CCW heat exchanger; the analysis concludes that no safety-related equipment is affected by the flooding (all PRA equipment in the CCW area is safety-related).

For large breaks in CCW piping, the maximum flood depth is 2.0 feet, which is bounded by flooding from ICW piping. If the entire volume of the CCW system flooded the CCW pit, the resulting depth of water would be less than 2.0 feet. For large breaks in ICW piping, the control room may receive low pressure alarms. In addition, the control room would receive a high level alarm from the CCW area sump. Another barrier to complete flooding is response time available. Before the CCW pit can start filling, the large pipe trench to the reactor auxiliary building (RAB) must first fill since it is at a lower elevation. Once the pit starts to fill (at a slower rate since it has a larger area), no PRA equipment is vulnerable until the level reaches at least the 17.2 foot level. It is unlikely that the level would reach this elevation since plant personnel enter the CCW area frequently. Since loss of PRA equipment due to flooding is not deemed credible in this scenario, it is screened from further consideration.

EPU Impact

Regarding flooding from a crack in the ICW system inlet line to the CCW heat exchanger, the EPU does not affect the ICW pump head performance or the ICW flow rates through the CCW heat exchangers. Regarding large breaks in CCW piping, there are no modifications to any CCW system equipment, components or system alignments resulting from the EPU, and therefore the EPU does not affect the volume of water in the CCW system. Regarding flooding from large breaks in ICW piping, the EPU does not affect the control room alarms for low pressure in the ICW system or high level in the CCW area sump. The EPU does not affect the locations of safety-related equipment in the CCW area. The EPU does not affect the frequency of plant personnel entering the CCW area. Therefore, the EPU does not affect the IPE assessment of the CCW area.

Condensate Pump and Condenser Area / Condensate Pump Pit

IPE Assessment

An analysis of this scenario suggests that a rupture of circulating water piping would cause operators to secure the circulating water pumps rapidly, and only a short gush of water would occur which would spread out over the area. However, if complete flooding of the pit is assumed, all three condensate pumps would be lost (this assumption is not based on calculated flood volume and therefore may be conservative). Flooding of all three condensate pumps bounds any spray concerns for this zone. The flood scenario could not be screened out qualitatively, therefore calculation of the contribution to CDF is discussed in the IPE submittal.

EPU Impact

The EPU does not affect the condensate pumps or their locations. The EPU does not affect the conservative assumption of complete flooding of the condensate pump pit due to a rupture of circulating water piping, resulting in the loss of all three condensate pumps. Therefore, the EPU does not affect the IPE assessment of the condensate pump and condenser area / condensate pump pit.

Feedwater Pumps 1A and 1B

IPE Assessment

The concern is damage to certain pressure transmitters (PTs) / flow transmitters (FTs), resulting in loss of one feedwater pump (the deluge could not affect PTs / FTs for both feedwater pumps at once). Several factors make this an incredible scenario. First, only a small class of pipe breaks would have the right size, geometry and location to deluge the PTs / FTs. Second, much of the heater drain fluid would flash to steam, reducing the amount of water available to impact the PTs / FTs. Third, the PTs / FTs are designed for exposure to the environment and it is unlikely that sufficient water would impinge upon the components to fail them. Even if this scenario was to occur, the loss of a single feedwater pump is not of great importance since the other feedwater pump, the condensate pumps, and the auxiliary feedwater pumps remain available for use. This scenario is therefore screened from further analysis.

EPU Impact

The feedwater pumps are being replaced for EPU, but their locations remain unchanged. Therefore, fluid from a pipe break at EPU conditions would not affect both feedwater pumps. As part of the feedwater replacement modification, the following changes affecting flow / pressure transmitters associated with each feedwater pump (1A / 1B) are being made: (1) the existing FT in the feedwater pump suction line is being replaced with an FT having a greater range, (2) a redundant FT in the feedwater pump suction line is being added for reliability and to replace an existing flow indicating switch (FIS), (3) two existing pressure switches (PSs) in the feedwater pump lube oil subsystem are being replaced with two PTs, and (4) two existing PSs in the feedwater pump suction line are being replaced with two PTs. All of the functions performed by the existing instrumentation are performed by the new instrumentation. The locations of the new instruments will be similar to the existing instruments. The new instruments will be designed for exposure to the environment, similar to the existing instruments. The heater drain pump discharge operating pressure and temperature at EPU conditions are not significantly different from the operating pressure and temperature at current conditions. Accordingly, the factors described in the IPE Assessment that make the scenario of damage to PTs / FTs causing loss of one feedwater pump to be not credible, are not affected by the EPU. Therefore, no initiating event will occur for this scenario, and the IPE assessment of feedwater pumps 1A and 1B is not affected by the EPU.

Aerated Waste Storage Tank (AWST) and Main Hallway East (El. 19.5 ft)

IPE Assessment

The equipment associated with an initiating event or PRA equipment located at the lowest elevation on the -0.5 foot elevation of the RAB are the boric acid makeup pumps, which are about a foot off the floor, and the charging pumps, which are about two feet off the floor. 0.38 feet of water will accumulate at the -0.5 foot elevation from release of the (full) aerated waste storage tank volume at the west end of the building. This level is not enough to damage any of the equipment associated with an initiating event or PRA equipment located on the -0.5 foot elevation. Release of the caustic storage tank contents is bounded by the AWST scenario. Release of the contents of the refueling water tank (RWT) via the pipe tunnel pathway is bounding for this scenario (water reaches a depth of 4.3 feet); refer to the analysis under "Pipe Tunnel," below.

EPU Impact

As addressed in Section 2.5.6.2.2 of the License Amendment Request (LAR) Attachment 5, no impact is expected on the system equipment in the liquid waste management system (LWMS). Therefore, the EPU does not affect the volume of the AWST in the LWMS system. The caustic (sodium hydroxide) storage tank in the containment spray system is not affected by the EPU. Therefore, the EPU does not affect the IPE assessment of the AWST and main hallway east (El. 19.5 ft).

Shutdown Heat Exchanger (SDHX) Rooms 1A and 1B

IPE Assessment

Since the plant is assumed initially at full power, the worst case break is that of a CCW line. CCW would be lost from one train, flooding that train's room. Assumed flooding of either room bounds any spray damage concerns. The only equipment associated with an initiating event or PRA equipment in either room is the equipment associated with that train's operation. The room associated with one train is separated from the room associated with the other train by a wall seven feet high. Each SDHX room has approximately 26,000 gallons of volume available before the wall to the next room would be topped. The total CCW system volume is estimated to be about 78,000 gallons, which is three times one room's volume. However, several factors inhibit flooding of this area and make topping the dividing wall improbable. First, it is unlikely that the full CCW system volume would enter the room. Second, each room has a drain to the ECCS pump rooms, which would allow adequate drainage for smaller leaks and reduce flood volume for larger leaks. Third, the control room would be alerted to the problem from numerous alarms; the CCW surge tank has control room alarms for low level, and the ECCS pump room sumps each have high and high-high level alarms in the control room. Finally, the RAB hallway outside the SDHX rooms is frequently traveled and has a plant operator passing through it periodically. It is unlikely that seepage from below the door to the RAB hallway would go unnoticed for an extended period of time. In any event, the CCW leak would likely be terminated before 26,000 gallons escaped. Since only one CCW train is assumed affected, there is no initiating event for this scenario and it is screened from further analysis.

EPU Impact

The EPU does not affect the structural configuration of SDHX rooms 1A and 1B. The EPU does not affect the CCW pump head performance. As addressed in Section 2.5.1.1.2.2.2 of LAR Attachment 5, the EPU does not affect the design of the equipment and floor drains system in the RAB, which includes the system components in the SDHX rooms and ECCS pump rooms. The EPU does not affect the CCW surge tank level instrumentation or alarms. The EPU does not affect the frequency of plant personnel passing through the hallway outside the SDHX rooms. The EPU does not affect the assumption that only one CCW train would be affected. Therefore, the EPU does not affect the IPE assessment of SDHX rooms 1A and 1B.

Pipe Tunnel

IPE Assessment

A flood of the pipe tunnel is assumed to immediately flood the RAB open areas, since there are doors that open outward from the pipe tunnel. Several factors make early detection and mitigation of such flooding highly probable. First, the RAB hallway is frequently traveled. Second, drainage to the 1100 gallon emergency core cooling system (ECCS) pump room sumps would result in high and high-high level alarms in the control room. Third, source-specific alarms and operator actions are likely whether the source is the refueling water tank (RWT) or CCW. If the RWT was filled to the high-high alarm level, only about 136,000 gallons could spill before a low level alarm would sound in the control room. This would certainly alert operators to the source of flooding, and flooding is easily terminated by closing the RWT outlet motor operated valves (MOVs). If this operator action is not taken, the scenario is still screened out based on the final level after continued flooding of the RAB until the RWT level drops to the recirculation actuation setpoint, where its outlet MOVs automatically close. If the operators additionally fail to close the isolation valves in the drain lines to the sumps in the ECCS rooms, as directed by plant procedures, any resultant flooding of the ECCS pump rooms is bounded by the analysis under "ECCS Pump Rooms 1A and 1B," below. If the operators do close these valves, 4.3 feet of water could accumulate on the -0.5 foot elevation and would result in failure of the boric acid makeup pumps and charging pumps. However, no initiating event occurs as a result of this scenario and it is screened from further analysis. Flooding from CCW in the pipe tunnel is considered bounded by the analysis of flooding from the RWT, described above, since CCW is a closed system containing only about 78,000 gallons. Also, timely identification and termination of CCW leakage is supported by surge tank low level alarms and off-normal procedures for loss of CCW.

EPU Impact

The EPU does not affect the frequency of plant personnel passing through the RAB hallway. As indicated in the evaluation under "SDHX Rooms 1A and 1B" above, the EPU does not affect the design of the equipment and floor drains system in the RAB, which includes the system components in the ECCS pump rooms. The EPU does not affect the RWT high-high and low level alarms. The EPU does not affect the RWT recirculation actuation setpoint (RAS). The EPU does not affect the locations of the boric acid makeup pumps and charging pumps. There are no modifications to any CCW system equipment, components or system alignments resulting from the EPU, and therefore the EPU does not affect the volume of water in the CCW system. As indicated in the analysis under "SDHX Rooms 1A and 1B" above, the EPU does not affect the CCW surge tank level instrumentation or alarms. Therefore, the EPU does not affect the IPE assessment of the pipe tunnel.

ECCS Pump Rooms 1A and 1B

IPE Assessment

Early detection and mitigation are highly probable, since each train's room has a sump with high and high-high level alarms in the control room. (The sumps are only 1100 gallon capacity, so they would fill quickly.) Further, if the RWT was filled to the high-high alarm level, only about 136,000 gallons could spill before a low level alarm would sound in the control room. This would certainly alert operators to the source of flooding, and flooding is easily terminated by closing the RWT outlet MOVs. If this operator action is not taken, the scenario is still screened out based on the final level following continued flooding of an ECCS pump room until the RWT level drops to the recirculation actuation setpoint, where its outlet MOVs automatically close. One of two things could happen. First, the water could be contained in the room due to the watertight doors installed. (The room floor is at the -10 foot elevation; the doors are at the -0.5 foot elevation.) The room would flood to the 12.8 foot elevation. Since the heating, ventilation, and air conditioning (HVAC) ductwork is at the 15 foot elevation, transport of the water via HVAC openings is not a concern. This scenario would result in the failure of all high pressure safety injection (HPSI), low pressure safety injection (LPSI), and containment spray pumps, but would cause no initiating event. The other possibility is failure of the watertight doors or other penetrations to contain the water, resulting in flooding of the -0.5 foot elevation of the RAB. The resulting water depth on the -0.5 foot elevation would be about 2.1 feet. This water level would damage the boric acid makeup pumps and charging pumps, in addition to all HPSI, LPSI, and containment spray pumps. Again, there would be no initiating event. Since there is no initiating event in either scenario, both are screened from further analysis.

EPU Impact

As indicated in the evaluation under "SDHX Rooms 1A and 1B" above, the EPU does not affect the design of the equipment and floor drains system in the RAB, which includes the system components in the ECCS pump rooms. As indicated in the evaluation under "Pipe Tunnel" above, the EPU does not affect the RWT high-high and low level alarms, and does not affect the RWT recirculation actuation setpoint. The EPU does not affect the structural configuration of ECCS pump rooms 1A and 1B or the location of the HVAC ductwork in the rooms. The EPU does not affect the locations of the boric acid makeup, charging, HPSI, LPSI, and containment spray pumps. Therefore, the EPU does not affect the IPE assessment of ECCS pump rooms 1A and 1B.

Hold-up Tank Enclosure

IPE Assessment

This zone is sealed up to the 19.5 foot elevation where the walls separating the four tanks end and water could communicate between tank cubicles. There is a fire door at the 19.5 foot elevation, so all four cubicles would have to fill before spilling significant water into the RAB. There is a total volume of over 200,000 gallons available up to the 19.5 foot elevation. Since the combined volume of all 4 tanks is 160,000 gallons, no water could leave the hold-up tank rooms. No initiating event or PRA equipment would be damaged inside or outside this zone from a release equal to the combined volume of all 4 hold-up tanks, so the scenario is screened from further analysis.

EPU Impact

The EPU does not affect the structural configuration of the hold-up tank enclosure. As addressed in Section 2.5.6.2.2 of LAR Attachment 5, the hold-up tanks in the boron recovery system (which is part of the liquid waste management system) accumulate liquid waste discharges until sufficient amounts are ready for processing and have adequate capacity to handle waste flow surges during normal plant operation. There are no new sources of potentially radioactive liquids, and the EPU does not change the collection, segregation, processing, discharging, or recycling of radioactive liquid wastes. Accordingly, the EPU does not affect the combined volume of water in the hold-up tanks. Therefore, the EPU does not affect the IPE assessment of the hold-up tank enclosure.

"AB" Switchgear Room

IPE Assessment

Spray from overhead piping in the AB switchgear room could disable a motor generator (MG) set or the AB switchgear. A conservative assumption is made that a single spray source might affect both of these components. Loss of an MG set would trip the reactor. The AB switchgear powers the "C" ICW pump and "C" CCW pump, which are normally in standby. This scenario could not be screened out qualitatively, and therefore quantification of the contribution to CDF is discussed in the IPE submittal.

EPU Impact

The EPU does not affect the overhead piping in the AB switchgear room. The EPU does not affect the conservative assumption that a single spray source might affect both an MG set and the AB switchgear. Therefore, the EPU does not affect the IPE assessment of the "AB" switchgear room.

"B" Switchgear Room

IPE Assessment

Flooding to a depth to cause an initiating event or PRA equipment damage is not deemed credible due to fire pump auto start alarms in the control room, which would prompt investigation, and the vigilance of plant operators (and other personnel) who pass through this room periodically. However, spray could impact components in this room. Spray could affect the 1B5 motor control center (MCC), but no initiating event would result. Therefore, spray is not considered further for this room.

EPU Impact

The EPU does not affect the frequency of plant personnel passing through the "B" Switchgear Room. The EPU does not affect fire pump auto start alarms in the control room. The only change to the 1B5 MCC due to the EPU is relocation of the power for the Unit 1 RAB ventilation fan motor 4B from this MCC to load center bus 1B1. Therefore, no initiating event will result due to spray damage to this MCC, and the EPU does not affect the IPE assessment of the "B" switchgear room.

Cable Spread Switchgear Room

IPE Assessment

Flooding to a depth to cause an initiating event or PRA equipment damage is not deemed credible due to fire pump auto start alarms in the control room which would prompt investigation, and the vigilance of plant operators (and other personnel) who pass through this room periodically. However, spray could affect components in this room. Spray could damage the Vital AC Bus; in the plant model loss of this bus affects turbine runback, the feedwater control system, and steam dump to the condenser. It is conservatively assumed that a reactor trip would occur due to loss of feedwater. This scenario could not be screened out qualitatively; quantification of the contribution to CDF is discussed in the IPE submittal.

EPU Impact

The EPU does not affect the fire pump auto start alarms in the control room. The EPU does not affect the frequency of plant personnel passing through the cable spread switchgear room. The EPU does not affect the assumption that spray from fire protection piping in this room could damage the Vital AC bus. Therefore, the EPU does not affect the IPE assessment of the cable spread switchgear room.

"A" Switchgear Room

IPE Assessment

Flooding resulting from the eyewash station piping to a depth to cause an initiating event or PRA equipment damage is not deemed credible due to the small diameter and low pressure of the source, a drain near the source, and the vigilance of plant operators who pass through this room periodically. Therefore this scenario is screened from further analysis.

EPU Impact

The EPU does not affect the design or operating conditions of the eyewash station piping in the "A" switchgear room, or the provisions for drainage in the room. The EPU does not affect the frequency of plant personnel passing through the room. Therefore, the EPU does not affect the IPE assessment of the "A" switchgear room.

Resin Addition Tank Area

IPE Assessment

Flooding from broken or open piping in this area (43 foot elevation of RAB) would find its way to the -0.5 foot elevation of the RAB via drains and the stairwell. Due to the high visibility of water falling down the stairwell and the vigilance of plant operators, such a leak would be discovered and terminated in a timely manner. Flooding from the boric acid (BA) batch tank is limited to its volume, 636 gallons. 0.03 feet of water would accumulate on the floor if the entire volume of the BA batch tank were released. No initiating event or PRA equipment damage results from this small quantity of water, so the scenario is screened from further analysis.

EPU Impact

The EPU does not affect the vigilance of plant operators detecting water falling down the stairwell from the 43 foot elevation of the RAB. The EPU does not affect the volume of water in the BA batch tank. Therefore, the EPU does not affect the IPE assessment of the resin addition tank area.

Control Room

IPE Assessment

The flow rate from a failure of the sink/washroom plumbing would be minimal due to the small size of the piping. It is not deemed credible that a quantity of water sufficient to result in component damage would accumulate in the control room; the control room is continuously occupied, so any leakage or flooding would be quickly identified and rectified before damage occurred to PRA or initiating event equipment. This scenario is screened from further analysis.

EPU Impact

The EPU does not affect the design of the sink/washroom plumbing in the control room. The EPU does not affect the conclusion that control room operators would quickly identify and rectify any leakage or flooding in the room. Therefore, the EPU does not affect the IPE assessment of the control room.

Component Cooling Water Surge Tank Room

IPE Assessment

Timely identification of CCW leakage is supported by CCW surge tank low level alarms. The surge tank has a volume of only 2000 gallons, so level would drop quickly on a leak too large for floor drains to accommodate. There are control room alarms for CCW surge tank low level. If it is assumed that doors which only have a latch keeping them shut against the force of flood water immediately open and pass the flood water, no substantial flooding could occur since the door from the CCW surge tank room opens to the roof. More importantly, since each surge tank is baffled to ensure that a failure of one train will not disable the redundant train, the worst case scenario is loss of a single train of CCW. It is assumed that this scenario would result in a controlled, manual shutdown; thus there is no initiating event and the scenario is screened from further analysis.

EPU Impact

The EPU does not affect the CCW surge tank location, capacity, or design. The EPU does not affect the CCW surge tank level instrumentation or alarms. The EPU does not affect the assumption that, in the event of the worst case scenario of loss of a single train of CCW, the scenario would result in a controlled, manual shutdown. Therefore, the EPU does not affect the IPE assessment of the CCW surge tank room.

Conclusion

Based on the evaluations above of the impact of the EPU on the discussions / results of the IPE assessment of the applicable flood zones, it is concluded that no new initiating events will result due to the EPU, and that the assessments of internal flooding in the IPE submittal are not affected by the EPU.

SBPB-2: 2.5.1.1.2-01, Equipment and Floor Drains
The NRC's acceptance criteria for the equipment and floor drains (EFDS) are based on GDC-2 and GDC-4, in which they require the EFDS to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures).

The licensee indicated in the LAR that the EPU will not impact the current seismic design of components in the EFDS, nor add any new equipment that would increase quantity of liquids or cause an inadvertent transfer of contaminated fluids to an uncontaminated drainage system. However, the licensee does not provide further information in the LAR to justify its analysis of the EPU effects on the EFDS.

Provide additional information regarding the impact of the EPU on the EFDS to verify that the effects remain bounded by the current design capability of the EFDS.

Response to SBPB-2

As stated in Section 2.5.1.1.1.1 of LAR Attachment 5, the regulatory requirement for internal flooding design review is limited to the effects of a postulated fire main pipe rupture, as presented in UFSAR Appendix 9.5A, Section 3.1.3. Water from a ruptured fire line could eventually drain toward the emergency core cooling system (ECCS) pump room sumps located at elevation -10 ft. The "ECCS Pump Room Flooding Analysis" presented in UFSAR Appendix 9.5A, Section 3.1.3, demonstrates that flooding will not result in a loss of ECCS function. Design features of the equipment and floor drains system in support of the UFSAR analysis include the following:

- The sump in each ECCS pump room (A / B) has a capacity of 1100 gallons.
- The "A" sump receives drainage water from a single 4" line and a single 3" line, which have a total flow capacity of approximately 130 gpm. The "B" sump receives drainage water from a single 3" line and two 4" lines, which have a total flow capacity of 210 gpm.
- Each ECCS pump room sump contains seismic class I level switches which provides a high sump alarm and a hi-hi sump level alarm in the control room. These instruments are physically and electrically independent from one another.
- Each of the drain lines to the ECCS pump room sumps has double isolation valves. The piping and valves are seismically designed. All of the valves can be remotely controlled in the control room by a single switch. In addition, each valve can be individually closed by a locally operated manual handwheel. The primary method for limiting flooding in the ECCS pump rooms is to positively close these valves.

The EPU does not affect the design of fire protection system piping. The EPU does not affect the design of the equipment and floor drains system in the reactor auxiliary building, including the system components in the ECCS pump rooms. Therefore, the EPU does not affect the current design capability of the equipment and floor drains system to limit flooding in the ECCS pump rooms.

As addressed in Section 2.5.1.1.2.2 of LAR Attachment 5:

- The EPU does not affect the seismic design of components in the equipment and floor drains system

- The EPU does not add any new equipment or modify existing equipment (e.g., pumps, strainers) in the reactor building, reactor auxiliary building, or fuel handling building that would result in increasing the quantities of liquids currently entering the equipment and floor drains system. Therefore, the sizing of the existing equipment and floor drains system remains acceptable at EPU conditions.

- Based on the design of the equipment and floor drains system, the review of historical evaluations and inspections, and the current procedures in place to address flooding in the reactor auxiliary building, it was concluded that an event causing damage to safety-related equipment as a result of flooding related backflow through equipment and floor drainage system, as outlined in IE Notice 83-44, could not occur at St. Lucie Unit 1. The EPU does not add any new equipment or modify existing equipment within the equipment and floor drains system in the reactor auxiliary building, and therefore does not affect the conclusions of the review performed in response to IE Notice 83-44.

The EPU does not add any new equipment or modify existing equipment in the equipment and floor drains system in the reactor building, reactor auxiliary building, and fuel handling building, and therefore, the existing system design features that prevent inadvertent transfer of contaminated fluids to an uncontaminated drainage system are not affected by the EPU.

<u>SBPB-3: 2.5.1.2.2, Turbine Generator</u>

The NRC's acceptance criteria for the turbine generator are based on GDC-4, and relates to protection of systems, structures, and components (SSCs) important to safety from the effects of turbine missiles. The probability of turbine missile generation is minimized by providing a turbine overspeed protection system (with suitable redundancy) to maintain turbine speed within an acceptable range following certain transients. However, the EPU may increase the peak turbine speed by increasing the energy within the turbine system.

The licensee stated in the LAR that the two existing low pressure turbines will be replaced with new Siemens-supplied replacement turbines prior to the implementation of the EPU. The licensee also stated that the overall turbine control system will be upgraded by Westinghouse to improve reliability and maintainability. However, the licensee does not provide information of how the replacement Siemens rotors will comply with the Westinghouse-provided turbine control system to continue to meet its current design analysis for missile protection under EPU conditions.

Provide additional information regarding the compatibility of the new Siemens rotors with the Westinghouse turbine control system to justify its assessment of missile protection under EPU conditions.

__Response to SBPB-3__

A turbine missile analysis was performed in support of EPU which reflects replacement of the low pressure (LP) turbine rotors with the Siemens BB281-13.9m^2 design. Its conclusions are similar to the existing analysis in that the probability of a disk failure with casing penetration is significantly below the NRC's limit value.

The analytical approach employed for the turbine missile analysis is in compliance with Siemens Westinghouse Topical Report (TR)-TP-04124, "Missile Probability Analysis for the Siemens 13.9M^2 Retrofit Design of Low-Pressure Turbine by Siemens AG." The NRC issued a Safety Evaluation Report to Siemens Westinghouse Power Corporation (SWPC) (ML040930616) which addressed, in part, the ability to reference SWPC Technical Report CT-27332-NP Revision 2 in retrofit applications such as in the case of St. Lucie Unit 1. CT-27332-NP justifies external missile probabilities out to 100,000 hours in comparison to the NRC limit.

The Topical Report expresses the probability of an external missile (P_1) emanating from the turbine-generator by conservatively evaluating two distinct types of LP shrunk-on disk failures, namely:

1. failure at normal operating speed up to 120% of the rated speed P_r and

2. failure due to run-away overspeed greater than 120% of rated speed P_o for all LP disks as follows

$P_1 = P_r + P_o = \sum^N_{i=1} P^i_{2r} * P^i_{3r} + P_{1o}$

Where the above terms are defined as:

P_1 probability of an external missile

P_r probability of an external missile for speeds up to 120% of rated speed

P_o probability of an external missile for speeds greater than 120% of rated speed

N, i total and current number of disks

P^i_{2r} probability of disk # i burst up to 120% of rated speed due to stress corrosion crack growth to critical size

P^i_{3r} probability of casing penetration given a burst of the disk # i up to 120% of rated speed

P_{1o} probability of a run-away overspeed incident (>120% of rated speed) due to a system separation initiation event followed by a failure of overspeed protection system

The P_o term reduces to the P_{1o} term based on the conservative assumptions in the analysis that the probability of disk failure and probability of casing penetration are both equal to 1.0 for speeds greater than 120%

The derivation of this equation is presented in the Topical Report.

For the calculation of P_r, a Monte-Carlo simulation technique involving successive deterministic fracture mechanics calculations using randomly selected values of variables is used. As a failure condition, the brittle fracture mode is assumed. Selected random variables that are defined by finite element analysis are employed and a critical crack size is defined. Limits are based on the applicability limitation of linear-elastic fracture mechanics and do not represent an imminent burst condition. In the final summation $\sum^4{}_{i=1} P_r$ was determined to be approximately two orders of magnitude below the NRC limit value for up to 100,000 operating hours between disk inspections providing that no cracks are detected in the discs.

Due to the conservative assumptions inherent in the turbine missile analysis methodology in the Siemens analysis, the calculation of disk failure probability and casing penetration (P_{2r} and P_{3r}) is treated completely independently from the calculation of system separation probability and overspeed protection system failure probability (P_{1o}) as detailed in the Westinghouse Owners Group (WOG) analysis, WCAP-16501, "Extension of Turbine Valve Test Frequency Up to 6 Months for BB-296 Siemens Power Generation Turbines With Steam Chests, Rev.0".

Turbine speeds greater than 120% of rated speed can only be reached due to a total functional failure of the turbine overspeed protection system. As discussed in section 2.5.1.2.2.2.2 of the EPU LAR, the larger inertia of the replacement LP rotors outweighed other EPU effects such that the net effect is a 1% decrease in expected maximum turbine speed during a loss of load event. Specifically, the expected change in the inertia of the rotor system will be approximately 47%. Also, the efficiency improvements due to the HP and LP upgrades and increase in thermal power will result in an increase in the power output of the turbine of approximately 14.4%. The increase in the amount of energy during the trip delays and valve closing as well as expansion of the entrapped steam will be higher at the EPU condition. However, this increase is more than offset by the increased inertia of the rotor train. Thus, the calculated overspeed will be approximately 1% lower than the original equipment. The calculated overspeed upon a breaker opening at full power EPU conditions is 114.6% based upon a trip setpoint of 111%.

P_{1o} or the probability of a run-away overspeed incident (>120% of rated speed) is the product of the system separation probability and the overspeed protection system failure probability and it is based on Westinghouse Owners Group (WOG) analysis WCAP-16501. The existing overspeed protection system failure probability, P_{1o} documented in WCAP-16501 remains bounding for the upgraded turbine control system to be installed as part of EPU at St. Lucie Unit 1. The modifications being made to enhance the reliability of the controls system, and thereby reduce the probability of reaching a run-away overspeed condition, are not being credited in the failure probability analysis. Since the P_{1o} probability from WCAP-16501 is bounding for both the current and EPU turbine overspeed protection system St. Lucie designs, the existing 18 month fuel cycle conditional probability of destructive overspeed remains unchanged due to the power uprate. This WCAP was prepared for a number of participating plants including St Lucie, North Anna, McGuire, Waterford, Shearon Harris, Farley, Byron and Braidwood. Some of the participating plants, including Byron Units 1 & 2, had previously upgraded their turbine control and turbine protection systems based on a Westinghouse Ovation design similar to the proposed new St Lucie Unit 1 Ovation design for EPU.

The design of the Ovation turbine control and protection system at St. Lucie Unit 1 includes two independent subsystems each with redundant processors. Each subsystem provides turbine overspeed protection in 2 out of 3 logic. Examples of some of the new design features include: redundant and diverse speed probes, speed sensing modules independent of control processors, and a redundant and diverse Overspeed Protection Control (OPC) trip system at 103% overspeed. The Ovation turbine control and protection system enhances control system reliability and thereby reduces the probability of reaching a run-away overspeed condition. Thus, continued usage of the existing overspeed protection system failure probability in the overspeed analysis is conservative.

Total probability of an external turbine missile (P_1) for St. Lucie Unit 1 at 100,000 hours inspection interval is approximately 60 times less than the NRC limit value of 11.42 E-5 per 100,000 hours inspection interval. Therefore, it has been shown that the replacement Siemens rotors provided with a Westinghouse-provided Ovation control system continue to meet the current design analysis for missile protection under EPU conditions.

SBPB-4: 2.5.4.1-01, Spent Fuel Pool Cooling and Cleanup System

The NRC's acceptance criteria for the spent fuel pool cooling and cleanup (SFPCC) system are based on three regulatory factors: (1) GDC-5, which requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (2) GDC-44, which requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided; and (3) GDC-61, which requires that fuel storage systems be designed with residual heat removal (RHR) capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions.

The licensee stated in the LAR that an instantaneous full core offload 120 hours after shutdown, the spent fuel pool bulk temperature reached a maximum of approximately 125°F with two fuel pool cooling pumps in operation. The licensee later stated that during EPU conditions, a full core offload 140 hours after shutdown would produce a bulk temperature of less than 150°F in the spent fuel pool. The licensee does not provide justification for why the analysis was performed at 140 hours for EPU conditions as opposed to 120 hours for the current design capability of the spent fuel pool.

Provide additional information to justify the EPU analysis of full core offload 140 hours after shutdown for the spent fuel pool as opposed to the current analysis of 120 hours after shutdown.

Response to SBPB-4

For St. Lucie Unit 1, the current full core offload analysis, with two fuel pool cooling pumps in operation, supports a timing of 120 hours after shutdown for an instantaneous core offload to meet the 150°F criterion for the maximum spent fuel pool (SFP) bulk temperature. The calculated maximum SFP bulk temperature for this scenario was approximately 125°F. The current offload time requirement is however defined by the analysis with only one fuel pool cooling pump in operation. The current analysis, with one fuel pool cooling pump in operation, supports a timing of 168 hours after shutdown to initiate full core offload, unless an earlier time is justified based on cycle specific decay heat load calculation.

The initiation time for core offload has been greater than 140 hours for past several cycles and this time to initiate core offload is controlled via plant procedures as verified every cycle based on the cycle-specific fuel pool heat load. Since the actual timing to initiate the full core offload is well beyond 120 hours after shutdown, a less limiting but conservative time of 140 hours after shutdown, to initiate core offload, was chosen for the EPU analysis with one and two fuel pool cooling pumps in operation. Extending the core offload time to 140 hours after shutdown is also based on the fact that the fuel pool heat load, in the EPU analysis, is increased due to the following:

- Nominal operating power of discharged fuel is increased from 2700 MWt to 3020 MWt.
- All the fuel assemblies in the pool, including the pre-EPU fuel, are treated as being irradiated at EPU power.
- The number of fuel assemblies discharged per cycle is increased due to the increased fresh fuel requirement for EPU, which increased the heat load of the previous cycle discharged fuel.
- The decay time for the previous cycle discharged fuel is reduced to 15 months, which conservatively bounds typical 18 month cycle decay times.

SBPB-5: 2.5.4.3-01, Reactor Auxiliary Cooling Water Systems

The NRC's acceptance criteria for the reactor auxiliary cooling water system (or component cooling water (CCW) as described for St. Lucie) are based on three regulatory factors: (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and (3) GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

The licensee indicated in the LAR that calculations for potential for two-phase flow, void formation, and water hammer after EPU implementation were performed to compare each scenario to current design capability of the CCW. The licensee did not clearly state for each scenario how the EPU analysis was compared to the current design analysis to make the determination that the CCW will handle the water hammer effects of the EPU after a potential CCW restart.

Provide additional information to justify how the calculations for two-phase flow, void formation, and water hammer accounted for the EPU effects to determine that the current design analysis is still applicable after EPU implementation.

Response to SBPB-5

Generic Letter GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," postulates that the Component Cooling Water (CCW) system could experience the effects of concurrent Loss of Coolant Accident or Main Steam Line Break (LOCA or MSLB) with Loss of Offsite Power (LOOP).

If a combined LOCA/LOOP or MSLB/LOOP did occur, water in the fan coolers would heat up as the containment reaches post-LOCA temperatures and pressures. A portion of the water in the coolers could potentially boil and convert to steam. The low pressure at the coolers as a result of loss of pump head would aid in the boiling process by lowering the temperature at which boiling will occur. When the pumps are eventually started off the diesel generator bus signal, the column rejoining of the void created by steam generation may render the system susceptible to water hammer damage as the piping refills. To evaluate this, column closure water hammer (CCWH) loads are developed using the EPRI GL 96-06 Method of Characteristics (MOC) methodology.

Prior to performing the CCWH analysis, heat transfer analyses are performed to determine water temperatures and estimated void fraction in each of the Containment Fan Coolers (CFCs). Specifically, CFC water temperature time histories from containment pipe break environments are calculated. To aid in the heat transfer calculations, a coast down analysis following pump trip is performed using HYTRAN. CFC flow rates and pressures from the coast down analysis are used in determining the saturated steam pressure time history in the cavity as it grows in size.

In performing the CCWH analysis, HYTRAN is used to determine the growth and closure of a discrete steam and air filled cavity using the EPRI GL 96-06 MOC methodology (water hammer analysis). As the two columns on either side of the void close, water hammer loads are developed, and are calculated in HYTRAN. The void pressure time history along with an estimate of the amount of dissolved air released into the cavity during boiling are input to HYTRAN to define the boundary condition at the discrete cavity node. The amount of air release is calculated following the EPRI GL 96-06 MOC methodology.

Pre-EPU analysis concluded that a representative portion of the piping associated with the St. Lucie Component Cooling Water system meets ASME Code acceptability limits following a postulated GL 96-06 Column Closure Water Hammer (CCWH) event. Because of piping/support configuration and system performance similarities, it is concluded that these stresses and support loads are representative of the remainder of the Unit 1 and 2 CCW piping under CCWH loading.

The analytical methods used for St. Lucie were specifically reviewed by the NRC in the closure of GL 96-06. The NRC requested additional information on these methods via letter from Mr. Brendan T. Moroney to FPL's J. A. Stall "St. Lucie Plant, Unit 1 and 2 - Request for Additional Information Regarding the Resolution of Generic Letter 96-06 Waterhammer Issues," dated August 1, 2003 (ML032130002). FPL's response to the RAIs was submitted in a letter from William Jefferson, Jr. to the NRC, "Response to NRC Request for Additional Information Generic Letter 96-06 Waterhammer Issues," dated September 29, 2003 (ML032740069). NRC acceptance of FPL's response is documented in letter from Brendan T. Moroney to J. A. Stall (FPL), "Closeout of Responses to Generic Letter 96-06 Concerning Waterhammer and Two-Phase Flow for St. Lucie, Units 1 and 2," dated March 11, 2004 (ML040680741).

As a result of EPU, there is a minor change in containment pressure and temperature response. This affects the heat transfer to the CCW piping inside containment and the amount of boiling that may occur in the CCW piping when the pump is off. Void sizes are calculated for each of the coolers on both units. The void size calculation determines the approximate size of voids formed and the time of bulk boiling. The design basis void calculation concluded that the limiting case for void formation is that of a design basis accident LOCA/LOOP in cooler 1C. The EPU void calculation determined that the void for both coolers 1C (Unit 1) and 2C (Unit 2) is about the same size, which increases as a result of EPU. However, since the 1C and 2C coolers have equivalent size voids after EPU, and since the pipe routing and system design for both units are equivalent, the limiting case for void formation following EPU remains that for the Unit 1C cooler. The CCWH was re-performed on the bounding, Unit 1C, cooler.

The EPU water hammer calculation concluded that increased EPU containment temperatures lead to slightly increased saturation pressures in the steam void. The higher steam pressures act to slightly reduce the pump water column velocity following pump restart, as well as the water column closing differential velocity. As a consequence of the reduced closing velocity, the EPU calculated water hammer loads are lower than those previously calculated.

Therefore, it is concluded that EPU has no impact on the design basis GL 96-06 results or conclusions.

<u>SBPB-6: 2.5.4.5-01, Auxiliary Feedwater</u>
Section 2.5.4.5.2.1 of Attachment 5 to the LAR and Section 10.5.2 of the St. Lucie Unit 1 FSAR describe that the AFW system consists of one greater than full flow capacity and two full flow capacity auxiliary feedwater pumps. Clarify what the term full capacity means in this context and address whether the definition was changed to support operation at EPU. Also, confirm that the assumed AFW system operation for the various design basis accident analyses and the calculated AFW success criteria for probabilistic analyses of corresponding initiating events was unchanged as a result of assumed change in thermal power supporting EPU. If confirmed, describe the extent that the power increase was accommodated by available margin as opposed to other changes, such as the proposed increase in the AFW initiation setpoint for low steam generator water level.

<u>Response to SBPB-6</u>

St. Lucie Unit 1 AFW system consists of two motor-driven (MD) pumps and one turbine-driven (TD) pump. As shown in the Unit 1 UFSAR, Table 10.1-1, the motor-driven pumps provide 325 gpm per pump when run at full flow capacity and the turbine-driven pump provides 600 gpm when run at full flow capacity. The term "full flow capacity" as used in the UFSAR is intended to reflect the required flow to one steam generator as provided by a single MD AFW pump. Since the TD AFW pump provides flow in excess of a single MD AFW pump, the TD AFW pump is said to have "greater than full flow capacity."

For the purpose of PRA and associated EPU thermal hydraulic calculations, the flow capacity of the AFW pumps has not changed. As discussed in EPU LAR Attachment 5, Licensing Report (LR) Section 2.5.4.5, the flow capacity of the MD and TD AFW pumps continues to be adequate at EPU conditions. AFW system operation for the various design basis and PRA accident analyses remain unchanged as a result of the change in thermal power supporting EPU.

The calculated AFW success criteria for probabilistic analyses of corresponding initiating events as presented below are unchanged as a result of the change in thermal power supporting EPU. The existing design parameters of the MD and TD AFW pumps are adequate for EPU operation. The AFW pump flow rates and system provide the heat removal required for response to postulated accidents and cooldown modes under EPU conditions using the existing pumps, piping and valves. The increase in thermal power due to EPU does not diminish the ability of these pumps to provide full flow capacity in support of their design basis functions.

The increase in thermal power requires additional water inventory be available in the event AFW system operation is required. The condensate storage tank (CST) contains the water inventory necessary to support AFW system operation. A change to Technical Specification (TS) 3 / 4.7.1.3 "Plant Systems-Condensate Storage Tank," is proposed as presented in EPU LAR Attachment 1, item 22. This TS change revises the CST minimum water inventory to ensure that sufficient water is available to accommodate decay heat removal. In postulated accident scenarios, the operation of either AFW pumps or main feedwater pumps (MFWPs) are available to support the secondary heat removal safety function consistent with the success criteria table provided below. AFW pump success criteria does not change between current and EPU calculations.

AFW Success Criteria in Support of Secondary Heat Removal in Various Accident Scenarios.	
Accident Scenario	**Success Criteria**
Reactor Trip at SG Low level	1/3 AFW pumps to 1/2 SGs, or 1/2 MFWP to 1/2 SGs.
Reactor Trip at SG Normal Level	1/3 AFW pumps to 1/2 SGs, or 1/2 MFWP to 1/2 SGs.
ATWS	2 MD AFWP or one TD AFWP to 2/2 SGs, or 2/2 MFWP to 2/2 SGs.
Small LOCA	1/3 AFW pumps to 1/2 SGs, or 1/2 MFWP to 1/2 SGs.
Induced LOCA (e.g., RCP Seal LOCA, PORV LOCA)	1/3 AFW pumps to 1/2 SGs, or 1/2 MFWP to 1/2 SGs.
SGTR	1/3 AFW pumps to 1/2 SGs, or 1/2 MFWP to 1/2 SGs.
Intersystem LOCA	1/3 AFW pumps to 1/2 SGs, or 1/2 MFWP to 1/2 SGs.

As a point of clarification, it is noted that the AFW initiation setpoint for low steam generator water level is not increasing as presented in the request for additional information. However, as discussed in EPU LAR Attachment 5, LR Section 2.13.2.6.1.3, the setpoint for reactor trip at steam generator low level is changing. This setpoint change is considered a mitigating strategy that will be adopted during EPU operations when the AFW and MFW systems are assumed to be lost in order to offset the potential increase in risk due to a Total Loss of Feedwater (TLOFW) event at higher thermal power. This strategy has no impact on AFW system success criteria. As mentioned above, the AFW system is assumed to be lost.

<u>SBPB-7: 2.12-01, Power Ascension and Testing Plan</u>

The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service.

The licensee indicated in the LAR that equipment modifications are being made to the feedwater system to support EPU conditions and that the feedwater system response would be monitored during power ascension. The licensee also stated that the transient

response for the feedwater system during EPU conditions was modeled using the CENTS computer code. However, the staff is concerned that CENTS-modeled transient for the feedwater system may not reflect the actual transient response of the replacement feedwater pumps, along with its compatibility with the modified feedwater control system, during EPU conditions.

Discuss how the feedwater system will be assessed with actual transient testing prior to EPU implementation to confirm that the replacement feedwater pumps and feedwater control system will respond in a manner similar as the CENTS-modeled transient, or if actual transient testing will not be done, justify why it is not needed.

<u>Response to SBPB-7</u>

EPU License Amendment Request (LAR), Attachment 5, Section 2.12.1.2.6, discusses the CENTS code model and its general application. LAR Section 2.12.1.2.7 includes the application of CENTS to load transients. The following discussion provides supplemental information regarding the CENTS modeling and the justification for why actual transient testing is not required, with emphasis on the Feedwater (FW) System.

CENTS was benchmarked with operating data in order to refine the EPU model. The following St. Lucie transients were used in the benchmarking process:

- Unit 1 100% power automatic reactor trip from reactor coolant pump (RCP) 1A2 trip on June 5, 2001;
- Unit 1 100% to 68% power ramp on August 20, 2008;
- Unit 1 42% to 83% power ramp on March 14, 2008;
- Unit 2 manual reactor trip from 100% power on June 4, 2008 following a loss of a main FW pump; and
- Unit 2 manual reactor trip from 100% power on June 7, 2008 following a condensate pump trip.

The Unit 2 events were chosen to supplement the Unit 1 events based on their recent history, high power level, available plant data, and the fact that Unit 2 is sufficiently close to Unit 1 to support validation of the CENTS model. For benchmarking against each of the 2008 events, short duration time interval data for a large number of key plant parameters was available for electronic comparison.

A CENTS model base deck was developed taking into account EPU conditions. The model was built to incorporate the applicable EPU equipment modifications and setpoint changes as well as the EPU operating conditions. The EPU FW modifications modeled in the CENTS transient analyses include the valve Cv for the modified FW flow control valves, the pump curve for the new FW pumps, and the new FW control system settings. These modifications do not change the fundamental design and operation of the FW system. For the flow control valves, the valve trim is new, but the valve body is not replaced. For the new pumps, the internals are new, but the motor is not replaced. For the FW control system, Distributed Control System (DCS) software testing will include validation that the FW control system is correctly modeled in the CENTS code.

The agreement between the CENTS model results and the actual plant response during the benchmarking process, and the fact that the FW equipment systems have been appropriately incorporated in the CENTS model, provides confidence that CENTS cases adequately model plant response at EPU conditions.

The CENTS cases modeled the response to many parameters. LAR Attachment 5 Table 2.12-4, lists the CENTS cases. Each case provided results for the following parameters: reactor power, reactor coolant system (RCS) temperatures, pressurizer pressure, pressurizer level, steam generator (SG) pressure, SG level, SG steam flow, SG FW flow, steam header pressure, main FW valve position, bypass FW valve position, FW pump pressures (suction and discharge), FW temperature. Many of these parameters are related to the FW system, and the parameter responses demonstrated acceptable FW system response.

Although step change load transients are not planned, FW system response is monitored during power ascension. The monitoring includes evolutions such as swap from FW bypass to S/G flow control valves, second condensate pump start, second FW pump start, closure of second FW pump recirculation valve, and heater drain (HD) pumps start. These evolutions are FW system transients even though reactor power and load are not expected to change.

SBPB-8: 2.5.4.1-02, Spent Fuel Pool Cooling and Cleanup System

The NRC's acceptance criteria for the spent fuel pool cooling and cleanup (SFPCC) system are based on three regulatory factors: (1) GDC-5, which requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; (2) GDC-44, which requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided; and (3) GDC-61, which requires that fuel storage systems be designed with residual heat removal (RHR) capability reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions.

The licensee stated in Section 2.5.4.1 of the LAR that methods similar to what is described in the UFSAR Section 9.1.3.4 were used to perform spent fuel pool cooling analyses at EPU conditions. One of the assumptions described in the LAR for minimizing the heat transfer from the fuel pool is that the ambient air temperature in the vicinity of the fuel pool is assumed to be 110°F to minimize passive heat losses. However, in UFSAR Section 9.1.3.4, the licensee stated that the passive heat losses from the spent fuel pool surface to the Fuel Handling Building air assume the relative humidity of the building air is 100%. The LAR appears to introduce a revision to the spent fuel pool cooling analysis for St. Lucie Unit 1.

The staff requests that the licensee explain in detail how the methods used to perform spent fuel pool cooling analyses were changed. If new methods were introduced, such as modeling heat loss to the structure or environment, the staff also requests that the licensee provide the model and associated benchmarking or validation completed to establish the quality of the model.

Response to SBPB-8

The current licensing-basis spent fuel pool cooling calculations were performed in 2001-2003, when the allowance for installing a spent fuel storage rack in the cask pit was requested (Unit 1 OL Amendment 192). These methods used in these calculations were summarized in Chapter 5 of a Holtec-prepared Licensing Report HI-2022882, which was submitted to NRC for review as part of the OL Amendment package (Enclosure to FPL Document L-2002-187, dated 23 October 2002). The NRC approved OL Amendment 192 via letter with SER on 9 July 2004 (ADAMS Accession Number ML041960439).
The EPU spent fuel pool cooling calculations were performed using the same analysis methods that underlie the current licensing basis. The following table compares the methods and tools used to perform the two sets of spent fuel pool cooling calculations:

Method Feature	Current Licensing Basis	EPU
Fuel Decay Heat Calculations		
Decay Heat Method	ORIGEN2	ORIGEN2
Analysis Programs	LONGOR BULKTEM	LONGOR BULKTEM
Bulk Temperature Calculations		
Steady-State or Transient	Transient	Transient
Analysis Program	BULKTEM	BULKTEM
Consider Passive Heat Loss	Yes, assuming 100% R.H.	Yes, assuming 100% R.H.
Time-to-Boil Calculations		
Analysis Program	TBOIL	TBOIL
Consider Passive Heat Loss	Yes, assuming 100% R.H.	Yes, assuming 100% R.H.
Local Temperature Calculations		
Analysis Method	CFD	CFD
Steady-State or Transient	Steady-State	Steady-State
Analysis Program	FLUENT	FLUENT
Consider Passive Heat Loss	No	No

As the above table shows, compared to the current licensing basis, none of the methods used to perform spent fuel pool cooling analyses were changed in the EPU calculations.

Noting that the RAI implies that the use of passive heat losses is a change from the current licensing-basis methodology, it is worth pointing out that Subsection 9.1.3.4.2.2.1 of the current St. Lucie Unit 1 UFSAR includes a passive heat loss term in the governing equation for the pool bulk temperature response, and states:

> "The passive heat loss to the environment includes conduction heat transfer through the SFP walls and slab as well as natural convection, thermal radiation and mass dilution from the surface of the SFP water."

The passive heat loss formulation used for the EPU analyses is the exact same one used in the current licensing basis analyses and includes all of these same heat and mass transfer mechanisms. Further, this subsection of the UFSAR also indicates that a relative humidity of 100% is part of the current licensing basis:

> "The passive heat losses from the SFP surface to the FHB air assumes the relative humidity of the building air is 100%."

The above discussion shows that the spent fuel pool cooling calculations performed for the EPU use the identical methodology as the current licensing basis calculations.