CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-093
10CFR50.4

JUN 2 9 2001

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE STEAM GENERATOR REPLACEMENT
AND POWER UPRATE LICENSE AMENDMENT APPLICATIONS

Dear Sir or Madam:

By letters dated October 4, 2000 and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). NRC letter dated June 18, 2001 requested additional information to support staff review of the proposed license amendment requests. The requested information is provided by the Enclosure to this letter.

The enclosed information is provided as a supplement to our October 4, 2000 and December 14, 2000 submittals and does not change the purpose or scope of the submittals, nor does it affect the conclusions of either the no significant hazards considerations or environmental evaluations previously submitted.

Please refer any questions regarding the enclosed information to Mr. Mark Ellington at (919) 362-2057.

P.O. Box 165
New Hill, NC 27562

T> 919.362.2502
F> 919.362.2095

A001

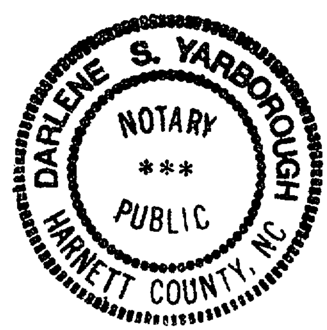
Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Darlene S. Yarborough
Notary (Seal)




My commission Expires: 2-21-2005

KWS/kws

Enclosure

c: Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. N. Kalyanam, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

bc:
Ms. D. B. Alexander
Mr. G. E. Attarian
Mr. L. R. Beller (BNP)
Mr. C. L. Burton
Mr. J. R. Caves
Mr. H. K. Chernoff (RNP)
Mr. W. F. Conway
Mr. G. W. Davis
Mr. J. W. Donahue
Mr. R. J. Duncan II
Mr. R. J. Field
Mr. W. J. Flanagan
Mr. K. N. Harris
Ms. L. N. Hartz
Mr. C. S. Hinnant
Mr. J. W. Holt
Mr. M. T. Janus
Mr. W. D. Johnson
Ms. T. A. Hardy (PE&RAS File)
Mr. R. D. Martin
Mr. T. C. Morton
Mr. W. M. Peavyhouse
Mr. J. M. Taylor
Nuclear Records
Harris Licensing File (s) (2 copies)

Response to Request for Additional Information
Regarding a License Amendment Request to Permit Uprated Power Operations
At the Shearon Harris Nuclear Power Plant

NRC Question 1

With regard to feedwater system, the steam generator replacement (SGR) and power uprate will require increases in flows and pressure from those required for the current steam generators at the current power level. In Page 2.2-2 of Enclosure 7 for October 4, 2000 letter, Carolina Power and Light Company (CP&L) stated that these changes, along with the piping changes in the supply to the new steam generators, require re-analyses of the system with respect to issues such as: water hammer potential due to rapid closure of the main feedwater isolation valve, and system pressure transients due to postulated transients such as bubble collapse. However, CP&L has not provided/discussed the results of this re-analyses. Please provide detailed discussions to address the effects of the SGR/Uprate on these issues.

CP&L Response

On page 2.2-2 of Enclosure 7, it is stated that these changes required reanalysis of the system with respect to issues such as water hammer and system pressure transients. This reanalysis was completed and the following is a summary of the results:

The Feedwater piping has been reanalyzed to account for changes in the configuration due to the SGR and the increased flow rates due to the Power Uprate. The piping configuration was changed due to the elimination of the pre-heater by-pass connection between the Auxiliary and Main Feedwater piping. In addition to this, the nozzle location of the Main Feedwater piping was raised in elevation, because the replacement steam generators have a feed ring design rather than the pre-heater design of the old steam generators.

The analysis performed maintained piping stresses below ASME code allowables. This was accomplished by either evaluation, reanalysis, and or modifications to ensure that the piping and supports were within code allowables.

Calculations were performed to analyze the Feedwater piping for water hammer and major fluid transient concerns with respect to feedline break (faulted) and non-break (upset) conditions with respect to rapid check valve closure (slam) and rapid Main Feedwater Isolation Valve (MFIV) closure. These calculations determined the time dependent pipe reaction forces as a result of the piping changes and the power uprate flow conditions. Piping, supports, penetration anchors, and equipment nozzles were qualified and determined to be acceptable for the revised loads.

In addition, other FW and AFW pipe breaks were analyzed, and the impacts of jet impingement, sub-compartment pressurization, and structural adequacy were evaluated. The evaluations concluded that the configurations were acceptable.

NRC Question 2

With regard to the adequacy of spent fuel pool (SFP) cooling during core off load, in Page 2.9-2 of Enclosure 7 for October 4, 2000 letter, CP&L stated that administrative controls are placed on the minimum cooling time (spent fuel assemblies (SFAs) "in-reactor" hold time) prior to transferring irradiated fuel from the core to the SFP in order to maintain the pool at less than or equal to 137°F[1] during core off load outages. Since the heat removal capability of the SFP cooling system is a function of the component cooling water system (CCWS) supply water temperature and the decay heat load is a function of the SFAs "in-reactor" hold time prior to discharge SFAs from the reactor, please provide the following information:

a. For the case of planned full core off load (referred to as normal full core offload shuffle in the FSAR) with a worse case single failure of the SFP cooling system, provide the calculated SFAs in-reactor hold time required for various CCWS supply water temperatures (i.e. 90°F, 95°F, 100°F, 105°F, 110°F, 115°F, 120°F, 125°F, etc.).

b. Demonstrate that the worst case of single failure of component/system has been identified.

c. Briefly discuss the provisions established in administrative controls to require analyses be performed to determine/establish SFAs "in-reactor" hold time required prior to discharge SFAs from the reactor to ensure that the SFP water temperature limit of 140°F will not be exceeded.

CP&L Response

[In a telecon with the NRC on 6/6/01, the request was modified. The revised request was to provide description of the inputs; methods and procedure used in determining the core offload wait time]

a. HNP Administrative Procedure PLP-114, " Relocated Tech Specs and Design Basis Requirements" contains the correlation between the CCW supply temperature and the minimum required wait time prior start of core offload. This procedure is subject to review by the Plant Nuclear Safety Committee (PNSC) and approved by the Plant General Manager. General Operating Procedure GP-009, "Refueling Cavity Fill, Refueling and Drain of the Refueling Cavity," implements the required hold time.

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1 This SFP water temperature limit of 137 °F will be increased to 140°F for SGR/Uprate operations.

The correlation is based on an analysis of the heat transfer capability of the Fuel Pool Cooling system as a function of CCW supply temperature and the decay heat in SFP A/B at completion of core unload as a function of time. The heat load input is comprised of the decay heat from the spent fuel previously located in Spent Fuel Pools A and B (SFP A/B) prior to the start of the refueling (base heat load) and the decay heat from the fuel in the reactor core to be discharged to the SFP A/B. The base heat load is assumed to be constant. The reactor core decay heat reduction as a function of time is calculated.

The heat transfer capability of the Fuel Pool Cooling system is calculated for a range of CCW supply temperatures, a fixed CCW flow rate and a fixed Fuel Pool Cooling flow rate. The two correlations (heat load versus time and heat removal capability versus CCW supply temperature) are combined to derive the final relationship between the hold time and CCW supply temperature. The correlation is truncated based on the minimum time after reactor shutdown assumed in radiological analysis (fuel handling accident).

b. The SRP 9.1.3 requires that the SFP cooling system be capable of withstanding a single active failure. FSAR Section 9.1.3.3 discusses the consequences of a single active failure. The only active components in the SFP cooling system are the SFP cooling pumps. The analysis of core offload times was prepared based on the operation of a single SFP cooling pump for the Incore Shuffle and the Full Core Offload Shuffle.

c. Please refer to response to subpart a. of this response.

NRC Question 3

On Page 2.9-3 of Enclosure 7 for October 4, 2000 letter, CP&L stated that each fuel pool heat exchanger has a design duty of 15.06 x $10^6$ Btu/hr. Also, in Page 2.6-3 of Enclosure 7, CP&L stated that based on CCWS evaluation results, the CCW supply temperature peaks at 124.8 °F. However, for the case of planned (SGR/Uprate) full core off load, the heat loads for SFP A/B and SFP C/D are 40.56 x $10^6$ Btu/hr and 1 x $10^6$ Btu/hr, respectively. It is not clear how the SFP water will be maintained to only reach a maximum equilibrium temperature of 140 °F.

Please provide detailed discussions to demonstrate that, for the case of a planned full core off load with the CCWS supply water temperature at 124.8 °F and a worse case of single failure of the SFP cooling system, the SFP water will be maintained to only reach a maximum equilibrium temperature of 140 °F. In addition, please provide a curve to show the calculated SFP water temperatures as a function of reactor shutdown time.

CP&L Response

The 124.8 °F temperature is a maximum value analyzed for the CCW mode " Shutdown @ 350°F" listed in Table 2.6.1 (Enclosure 7) of the October 4, 2000 submittal. As indicated in Table 2.6.1, several specific operating conditions were evaluated. The predicted maximum value for Refueling is lower than 124.8°F. As described in the response to Question 2.a above, the core offload wait time is a function of the actual CCW supply temperature that exists at the time of the refueling.

Allowable core offload wait times are based on the steady-state condition of the spent fuel pool temperature at the end of core offload. A curve of pool heat up as a function of core offload is not part of CP&L's method.

NRC Question 4

With regard to the SFP water temperature monitoring system, please provide the following information:

a. The set-point of the high water temperature alarm for the SFP.

b. Information supporting a determination that there is sufficient time for operators to intervene in order to ensure that the SFP water temperature limit of $140^0$F will not be exceeded.

c. Discuss what are the corrective actions (i.e. prohibit fuel handing, aligning other systems to provide SFP cooling, etc.) to be taken in the event of a high SFP water temperature alarm.

CP&L Response

a. The setpoint of the high water temperature alarm is 105°F.

b. There is sufficient time for operators to intervene, because the cooling systems have several alarms in the Main Control Room that provide an early indication of abnormal conditions in the systems that support Spent Fuel Pool (SFP) Cooling, the conservatisms in the analytical method, and the Defense in Depth strategy used in outage risk management.

The alarms provided in the Main Control Room to alert the operator of high and low pool water level and high temperature in the fuel pools include the SFP Cooling loop low flow, low CCW flow to the inservice SFP heat exchangers, SFP Hi temperature, and Fuel Pool Hi/Lo level alarms. A common Main Control Room alarm is provided for high SFP heat exchanger inlet and outlet temperature, Fuel Pool Cooling Pump Strainer $\Delta p$ high, and SFP cooling pump low discharger pressure.

Conservatism exists in the methods used to define the core offload wait time. For example, the thermal inertia of the spent fuel pool water is not considered. As a result, in past refueling outages, spent fuel pool temperatures on the order of 140°F have not been observed. The maximum spent fuel pool temperature observed in the last refueling outage (RFO 9) was recorded as approximately 106°F.

Outage risk management is an integral part of outage planning, outage implementation, and response to changing conditions in an outage. Plant administrative controls on the outage risk management are contained in administrative procedure OMP-003 "Outage Shutdown Risk Management." The controls include a defense in depth approach.

The minimum requirement for CCW and SFP cooling systems for Modes 5 and 6 are specified in OMP-003. The required CCW pump capability in refueling (water level greater than 23 feet above the Reactor Vessel Flange and Upper Internals Removed) is one pump functional and one pump available. For SFP cooling, the minimum requirement is one SFP cooling pump functional and the second pump capable of providing flow prior to boiling in the Spent Fuel Pool. The support for SFP cooling requires the same train CCW pump, CCW heat exchanger, Emergency Service Water (pump and header), and emergency power supply.

The minimum complement of equipment is reviewed as part of entry into a new outage "Configuration" and is displayed in prominent locations throughout the site. If the minimum complement of equipment is lost, the Superintendent – Shift Operations has the authority to stop work (such as fuel movement) and initiate corrective actions to ensure nuclear safety. The Shift Outage Manager is also required to initiate a Condition Report when minimum OMP-003 requirements are not met as a result of equipment failure.

c. The response to the high temperature alarm is to:

i. Check the status of the SFP Cooling Pump.

ii. Check the status of the CCW pump, if the running CCW pump has tripped, the Abnormal Operating procedure for loss of CCW will be followed.

iii. Place the standby SFP cooling pump in service if the running SFP cooling pump has tripped.

iv. Perform local investigation such as the status of the breakers for the SFP cooing pump, visually inspecting the SFP cooling pumps, and check for proper CCW flow to the SFP heat exchanger.

v. Periodically check the pool temperature until cooling can be restored.

NRC Question 5

In the unlikely event that there is a complete loss of SFP cooling capability, the SFP water temperature will rise and eventually will reach boiling temperature. Provide the time to boil (from the pool high temperature alarm caused by loss-of-pool cooling to boiling) and the boil-off rate (based on the SGR/Uprate heat load of 42.46 x $10^6$ Btu/hr from the unplanned full core off-load). Also, discuss sources and capacity of make-up water and the methods/systems (indicating system seismic design Category) used to provide the make-up water.

CP&L Response

The time to increase from 105°F (alarm set point) to 212°F is approximately 9.7 hours. This is calculated based on a temperature difference of 107°F and the thermal inertia of the Spent Fuel Pool A and B complex of 11.02 °F/hr (for the PUR abnormal core offload total decay heat in the SFP A/B complex of 42.5E+6 BTU/hr).

At 212°F, the boil off rate is approximately 90 gpm. This was calculated using the latent heat of vaporization at 14.7 psia of 970.1 BTU/lbm and the density of makeup water (62.35 lbm/cubic foot).

A discussion of the source of makeup water to the Spent Fuel Pools is provided in HNP FSAR Section 9.1.3.3. The normal source of makeup water to the pools is the Demineralized Water System (DWS). As described in FSAR Section 9.2.3.2, this system is non-safety related (refer also to FSAR Table 3.2.1-1). Another backup water source is the seismic Category 1 Refueling Water Storage Tank (RWST).

Outage Management Shutdown Risk administrative controls require minimum water levels in the DWS Tank and the RWST as part of the refueling Configuration. This includes the availability of equipment to transfer the water from the respective tanks to the Spent Fuel Pools.

The DWS transfer pumps have a capacity of 300 gpm at 455 ft (ref.: FSAR Table 9.2.3-1), which is sufficient to provide the required quantity of water.

The transfer of the RWST inventory to the Spent Fuel Pools relies on the Fuel Pool Purification Pumps or the Spent Fuel Pool skimmer pump. Both of these pumps and their respective piping are non-safety-related. The capacity of the Purification pump is 325 gpm at a head of 60 ft and the capacity of the Skimmer pump is 385 gpm at 210 feet (Ref.: FSAR Table 9.1.3-2).

It is acceptable to rely on these non-safety-related components because the Post Outage Full Core Offload Case (SRP 9.1.3 Abnormal case) does not require the consideration of a single failure in the cooling systems.

NRC Question 6

With regard to condensate inventory:

On Page 2.8.6 of Enclosure 7 for October 4, 2000 letter, CP&L stated that the minimum required inventory to satisfy the cooldown operation is 116,178 gallons and the available inventory is 238,000 gallons in the CST.

On Page 2.17.3-2 (indicating TS Bases Change) of Enclosure 7 for October 4, 2000 letter, CP&L stated that review of the SBO transient has determined that the total condensate required to cooldown the RCS during the 4 coping period hours after SBO has increased to approximately 112,200 gallons. The usable volume of CST is 238,000 gallons.

On Page 25 (discussing bases for proposed change) of Enclosure 1 for October 4, 2000 letter, CP&L stated that the existing TS minimum CST volume is 270,000 gallons, providing a usable volume of 235,000 gallons.

Please clarify the above differences.

a. With the existing TS minimum CST volume of 270,000 gallons, what is the available condensate inventory in the CST, 238,000 gallons or 235,000 gallons?

b. What is the minimum required condensate inventory to satisfy the cooldown operation, 116,178 gallons or 112,200 gallons for an SBO event?

CP&L Response

a) The TS minimum CST volume of 270,000 gallons, as stated in Enclosure 1, page E1-25 of our October 4, 2000 letter, discussing the proposed change to the CST inventory bases, is a contained volume, including un-usable inventory, various instrument uncertainties and margin. Only 238,000 gallons of that TS minimum CST volume, however, is safety–related usable inventory.

   This response clarifies the statements made in our October 4, 2000 license amendment request, in Enclosure 7, Sections 2.8.4 (page 2.8.6) and 2.17.3.4 (page 2.17.3-2). It also corrects the statements made in Enclosure 1, discussing the proposed change for the CST inventory bases (page E1-25) and Enclosure 6, Section 4.2.4.4.1 (page 4.2-7).

b) The minimum required condensate inventory to satisfy a 4-hour coping period and cooldown operation for an SBO event is 117,013 gallons. This inventory was re-calculated and corrects the inventory stated in our October 4, 2000 license amendment request, in Enclosure 7, Sections 2.8.4 (page 2.8.6) and 2.17.3.4 (page 2.17.3-2).

NRC Question 7

For LOCA and MSLB Containment Analyses, please indicate key input parameters that are different from UFSAR besides SGR/Power Uprate-related parameters and the effect on the peak containment pressure and temperature.

CP&L Response

The key input parameters for the LOCA and MSLB peak temperature and pressure Containment Analyses are noted in HNP FSAR Table 6.2.1-5 (plant initial conditions) and Table 6.2.1-6 (ESF Assumptions). These parameters are also discussed in various locations within FSAR Section 6.2.1 and Appendices 6.2B and 6.2C. The re-analyses to support Steam Generator Replacement (SGR) and Power Uprate (PUR) also included changes to these key inputs that are non-SGR / PUR related and are described below. These changes, by themselves, have NOT been specifically evaluated for their impact on Containment peak pressure and temperature, but were included in the SGR / PUR re-analyses. Unless otherwise stated, the parameters affect both LOCA and MSLB Containment re-analyses.

| PARAMETER | EXISTING FSAR VALUE | REVISED VALUE |
|---|---|---|
| **RCS Temperature Uncertainty** | + 4°F | + 6°F |
| **Containment Initial Pressure.** | 1.9 psig (for LOCA)<br>1.9 psig (for MSLB peak press)<br>-4"wg (for MSLB peak temperature) | 1.6 psig (LOCA)<br>1.6 psig (for MSLB peak press)<br>-1"wg (for MSLB peak temperature) |
| **Containment Initial Relative Humidity** | 65% RH – MSLB peak temperature | 75% RH – MSLB peak temperature |
| **SI Accumulator Nitrogen** | Not considered. | Nitrogen gas release to Containment accounted for during a LOCA. |
| **Containment Spray Flow** | 1832 gpm per safety train (LOCA)<br>1853 gpm max. per safety train. (MSLB). Partial spray credited per FSAR Tables 6.2B-1, 6.2B-2. | 1730 gpm per safety train. (LOCA).<br>1730 gpm per safety train. (MSLB) |
| **Containment Spray Start Time** | 57.27 sec (LOCA)<br>41.59 sec (MSLB - credits partial spray flow) | 58.4 sec (LOCA)<br>58.4 sec (MSLB) |

| | | |
|---|---|---|
| **Containment Heat Removal** | Assumes a service water flow rate of 1360 gpm per fan cooler. | Assumes a reduced service water flow rate of 1300 gpm per fan cooler. Also includes fan cooler motor heat load. |
| **RHR HX UA Overall Heat Transfer Coefficient (LOCA)** | $1.635 \times 10^6$ BTU /hr-°F. | $1.729 \times 10^6$ BTU /hr-°F near peak temperature. Otherwise analysis uses a variable UA depending on fluid temperatures. |
| **CCW Flow to RHR HX (LOCA)** | $2.80 \times 10^6$ lbm /hr | 4830 gpm |
| **Containment Sump Flow to RHR HX. (LOCA)** | $1.85 \times 10^6$ lbm /hr | 3700 gpm |
| **Feedwater Piping Volume SG to MFIV. (MSLB)** | 199 cu. ft | 245 cu. ft |
| **Main Steam Piping Volume SG to Turbine. (MSLB)** | 5751.5cu. ft | 9415 cu. ft |