CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-044
10CFR50.4
10CFR50.90

MAR 27 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
STEAM GENERATOR REPLACEMENT
LICENSE AMENDMENT REQUEST -
REVISED TECHNICAL SPECIFICATION
AND BASES PAGES

Dear Sir or Madam:

By letter HNP-00-142 dated October 4, 2000, CP&L submitted proposed Technical Specification (TS) changes related to the replacement of steam generators at the Harris Nuclear Plant (HNP).

The TS changes proposed by the October 4, 2000 submittal are primarily due to the physical differences between the original Westinghouse Model D4 pre-heat design steam generators (OSGs) and the replacement Westinghouse Model Delta 75 feedring design steam generators (RSGs). Setpoint-related changes, for example, are required due to the relocation of the water level instrumentation taps (nozzles) relative to the corresponding OSG tap locations. The corresponding setpoint-related changes and the basis for these changes are included in the October 4, 2000 submittal.

The increased tap-to-tap distance of the RSGs relative to the OSGs results in an increased calibrated span requirement for the steam generator level instrumentation. To meet this requirement, CP&L plans to replace the existing steam generator level transmitters with new Barton Model 764 transmitters. Replacing these level transmitters, however, will result in a change to one of the values in TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints. The October 4, 2000 submittal proposed multiple changes to TS Table 3.3-4, but the only proposed change affected by the transmitter replacement is the operability determination (Z) term associated with the Steam Generator Water Level – High-High (P-14) trip setpoint.

P.O. Box 165
New Hill, NC 27562

T> 919.362.2502
F> 919.362.2095

A001

The (Z) term of 9.63 proposed in the October 4, 2000 submittal would remain conservative for the new transmitters, because the uncertainties previously assumed in the calculation for the existing transmitters exceed those required for the new Barton Model 764 transmitters. A new (Z) term of 8.05 is now being proposed, however, to accurately reflect the value computed for the new transmitters.

Therefore, to address this change, a revised mark-up to TS page 3/4 3-32 is being provided as an enclosure to this letter. Please replace the TS page 3/4 3-32 mark-up provided in Enclosure 5 of the October 4, 2000 submittal with the mark-up of the same page enclosed herein.

In addition to TS page 3/4 3-32, mark-ups to TS page 5-6 and TS Bases page B 3/4 7-2 were also included in Enclosure 5 of the October 4, 2000 submittal. Since that time, however, TS page 5-6 has changed upon issuance of Amendment 103 and there has also been a change to TS Bases page B 3/4 7-2 (ref. HNP-01-014, dated February 14, 2001). Therefore, to address these changes, revised mark-up pages for both TS page 5-6 and TS Bases page B 3/4 7-2 are enclosed herein. Please replace the mark-ups of those same pages provided in Enclosure 5 of the October 4, 2000 submittal with the mark-ups enclosed herein.

The replacement pages enclosed herein do not affect the conclusions of either the 10 CFR 50.92 no significant hazards evaluation or the Environmental Evaluation previously submitted, nor do they alter or expand upon either the scope or purpose of the original submittal.

Please refer any questions regarding the enclosed information to Mr. Eric McCartney at (919) 362-2661.

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 CP&L.

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

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. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

bc:

Ms. D. B. Alexander
Mr. G. E. Attarian
Mr. R. H. Bazemore
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. W J. Hindman
Mr. C. S. Hinnant
Mr. J. W. Holt
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)

SHEARON HARRIS NUCLEAR POWER PLANT
NRC DOCKET NO. 50-400/LICENSE NO. NPF-63
STEAM GENERATOR REPLACEMENT
LICENSE AMENDMENT REQUEST
REVISED TECHNICAL SPECIFICATION PAGES 3/4 3-32, 5-6
AND BASES PAGE 3/4 7-2

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
b. Steam Generator Water Level--High-High (P-14)	~~15.0~~ 22.0	~~11.25~~ ~~9.63~~ 8.05	~~2.97~~ 2.0	≤ ~~82.4~~ 78.0% of narrow range instrument span.	≤ ~~84.3~~ 79.5% of narrow range instrument span.
c. Safety Injection	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low	~~19.2~~ 25.0	~~14.06~~ 16.85	~~2.97~~ 2.0	≥ ~~38.5~~ 25.0% of narrow range instrument span.	≥ ~~36.5~~ 23.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Trip Setpoint and Allowable Values.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation (Continued)					
b. Steam Generator Water Level--High-High (P-14)	22.0	8.05	2.0	≤ 78.0% of narrow range instrument span.	≤ 79.5% of narrow range instrument span.
c. Safety Injection	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level--Low-Low	25.0	16.85	2.0	≥ 25.0% of narrow range instrument span.	≥ 23.5% of narrow range instrument span.
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Trip Setpoint and Allowable Values.				
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.1.b is met. Delete — 1

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

approximately

Delete 5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of 580.8°F.

10,300

588.8

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.1.b is met.

CONTROL ROD ASSEMBLIES

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5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 10,300 cubic feet at a nominal T_{avg} of 588.8°F.

AUXILIARY FEEDWATER SYSTEM

maximum allowable — minimum

operation. The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15. The pump performance requirements are based upon a ~~4%~~ degradation of the ~~vendor~~ ~~certified~~ performance curves. Pump operation at this level has been demonstrated by calculation to deliver sufficient AFW flow to satisfy the accident analysis acceptance criteria.

delete — delete — pump

With regard to the periodic AFW valve position verification of Surveillance Requirement 4.7.1.2.1 Sub-paragraph b.1, this requirement does not include in its scope the AFW flow control valves inline from the AFW motor-driven pump discharge header to each steam generator when they are equipped with an auto-open feature. The auto-open logic feature is designed to automatically open these valves upon receipt of an Engineered Safety Features System AFW start signal. As a consequence, valves with an auto-open feature do not have a "correct position" which must be verified. The valves may be in any position, in any MODE of operation thereby allowing full use of the AFW System for activities such as to adjust steam generator water levels prior to and during plant start-up, as an alternate feedwater system during hot standby, for cooldown operations, and to establish and maintain wet layup conditions in the steam generators.

1 Delete

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for ~~12~~ hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics, and the value has also been adjusted in a manner similar to that for the RWST and BAT, as discussed on page B 3/4 1-3.

6

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F (a generic maximum) and are sufficient to prevent brittle fracture.

Insert E here

TS page B 3/4 7-2

INSERT E

The Shearon Harris specific RT_{NDT} is limited to a maximum value of 10°F.

AUXILIARY FEEDWATER SYSTEM

operation. The AFW System provides decay heat removal immediately following a station blackout event, and is required to mitigate the Loss of Normal Feedwater and Feedwater Line break accidents analyzed in FSAR Chapter 15. The minimum pump performance requirements are based upon a maximum allowable degradation of the pump performance curves. Pump operation at this level has been demonstrated by calculation to deliver sufficient AFW flow to satisfy the accident analysis acceptance criteria.

With regard to the periodic AFW valve position verification of Surveillance Requirement 4.7.1.2.1 Sub-paragraph b.1, this requirement does not include in its scope the AFW flow control valves inline from the AFW motor-driven pump discharge header to each steam generator when they are equipped with an auto-open feature. The auto-open logic feature is designed to automatically open these valves upon receipt of an Engineered Safety Features System AFW start signal. As a consequence, valves with an auto-open feature do not have a "correct position" which must be verified. The valves may be in any position, in any MODE of operation thereby allowing full use of the AFW System for activities such as to adjust steam generator water levels prior to and during plant start-up, as an alternate feedwater system during hot standby, for cooldown operations, and to establish and maintain wet layup conditions in the steam generators.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 6 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics, and the value has also been adjusted in a manner similar to that for the RWST and BAT, as discussed on page B 3/4 1-3.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F (a generic maximum) and are sufficient to prevent brittle fracture. The Shearon Harris specific RT_{NDT} is limited to a maximum value of 10°F.