Progress Energy

Christopher L. Burton
Vice President
Harris Nuclear Plant
Progress Energy Carolinas, Inc.

10 CFR 50.90

FEB 9 2012

Serial: HNP-12-002

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400 / RENEWED LICENSE NO. NPF-63

Subject: REQUEST FOR LICENSE AMENDMENT
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE
SUPPLEMENTAL INFORMATION

Reference: Letter from C.L. Burton to the U.S. NRC, "Request for License Amendment, Measurement Uncertainty Recapture Power Uprate," dated April 28, 2011.

Ladies and Gentlemen:

By letter dated April 28, 2011, Carolina Power & Light Company (CP&L), doing business as Progress Energy Carolinas, Inc., requested approval from the U.S. Nuclear Regulatory Commission (NRC) to increase the core thermal power level of Shearon Harris Nuclear Power Plant, Unit 1 (HNP) from 2,900 megawatts thermal (MWt) to 2,948 MWt, an increase of approximately 1.66 percent over the present licensed power level and to change the power plant technical specifications accordingly.

In a telephone discussion with the NRC staff, CP&L determined that it would be necessary to submit revisions to the previously submitted Technical Specification and draft Technical Specification Bases related to the implementation of TSTF-493. Those revisions are included in the enclosure to this letter. In addition, it is noted that applicable sections of procedure PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report," will be incorporated by reference into the FSAR prior to or concurrent with the implementation of the license amendment.

An additional non-technical change to the Technical Specification Section 3.3.2, Tables 3.3-1 and 4.3-1, revises the terminology from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure. This is further explained in the Enclosure, Attachment 4, to this letter.

The draft markups to the Technical Specification Bases are provided for information only.

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A001
NRR

CP&L originally requested license amendment approval by March 21, 2012, with implementation after Refueling Outage 17. Based upon the current schedule for the refueling outage, CP&L requests a 120-day implementation period for this amendment.

CP&L has concluded that the information provided in this supplement meets the intent of the original submittal and does not impact the conclusions of the Technical Analysis, No Significant Hazards Consideration, or Environmental Consideration as provided in the original submittal.

In accordance with 10 CFR 50.91(b), CP&L is providing the state of North Carolina with a copy of this response.

This document contains no new Regulatory Commitments.

Please refer any questions regarding this submittal to Mr. David Corlett, Supervisor – HNP Licensing/Regulatory Programs, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on
[FEB 8 2012].

Sincerely,

Christopher L. Burton

Enclosure: Harris Nuclear Plant Supplemental Information

cc: Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, Section Chief N.C. DENR
Ms. A. T. Billoch Colón, NRC Project Manager, HNP
Mr. V. M. McCree, NRC Regional Administrator, Region II

HNP-12-002

Enclosure

SHEARON HARRIS NUCLEAR POWER PLANT / UNIT NO. 1
DOCKET NO. 50-400 / RENEWED LICENSE NO. NPF-63

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE
TAC ME6169

HARRIS NUCLEAR PLANT
SUPPLEMENTAL INFORMATION

Attachment 1
Revised Technical Specification Mark-ups

Attachment 2
Revised Technical Specification Retyped Pages

Attachment 3
Revised Technical Specification Bases Markups

Attachment 4
Supporting Information Regarding Terminology Change from
Turbine Impulse Chamber Pressure to Turbine Inlet Pressure

HNP-12-002

Enclosure

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MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE
TAC ME6169

HARRIS NUCLEAR PLANT
SUPPLEMENTAL INFORMATION

Attachment 1
Revised Technical Specification Mark-ups
(10 pages plus cover)

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

2948

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of ~~2900~~ megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

DELETE

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. ~~134~~, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

---

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

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PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

2948

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2900~~ MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

DELETE

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

DELETE

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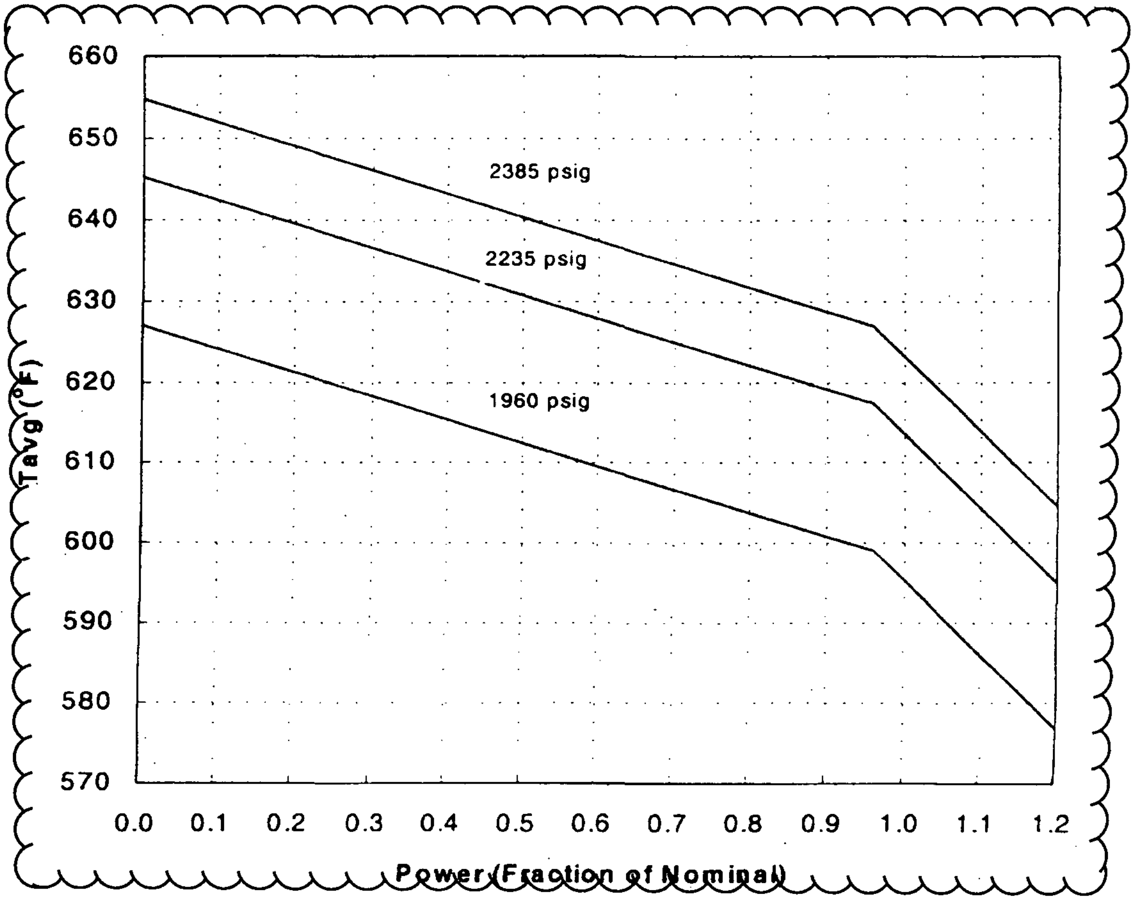

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW > [293,540 GPM X (1.0 + C_1)]

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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	~~7.5~~ 5.83	4.56	0	≤ ~~109~~ 108% of RTP**	≤ ~~111.1~~ 109.5% of RTP**
b. Low Setpoint	~~8.3~~ 7.83	4.56	0	≤ 25% of RTP**	≤ ~~27.1~~ 26.8% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	~~2.5~~ 2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	~~2.5~~ 2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10. Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11. Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

DELETE

**RTP = RATED THERMAL POWER

DELETE

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥ 1 x 10-10 amp	≥ 6 x 10-11 amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤ 10% of RTP**	≤ 12.1% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine ~~Impulse~~Inlet Pressure Equivalent	≤ 12.1% RTP** Turbine ~~Impulse~~Inlet Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤ 49% of RTP**	≤ 51.1% of RTP**
d. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	≥ 10% of RTP**	≥ 7.9% of RTP**
e. Turbine ~~Impulse Chamber~~Inlet Pressure, P-13	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine ~~Impulse~~Inlet Pressure Equivalent	≤ 12.1% RTP** Turbine ~~Impulse~~Inlet Pressure Equivalent
20. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.
22. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6	=	0.002/°F for T > T" and K_6 = 0 for T ≤ T".
T	=	As defined in Note 1.
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤588.8°F).
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

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NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Underfrequency Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6
17. Turbine Trip (Above P-7)					
a .Low Fluid Oil Pressure	3	2	2	1	6
b. Turbine Throttle Valve Closure	4	4	1	1	10
18. Safety Injection Input from ESF	2	1	2	1, 2	13
19, Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1	7
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
e. Turbine ~~Impulse Chamber~~Inlet Pressure, P-13	2	1	2	1	7

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19 Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
e. Turbine ~~Impulse Chamber~~Inlet Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
20 Reactor Trip Breaker	N.A.	N.A.	N.A.	H (7, 9, 10)	N.A.	1, 2, 3*, 4*, 5*
21 Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	H (7)	1, 2, 3*, 4*, 5*
22 Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	H (7, 13, 14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	~~50~~ 49
2	~~33~~ 32
3	~~16~~ 15

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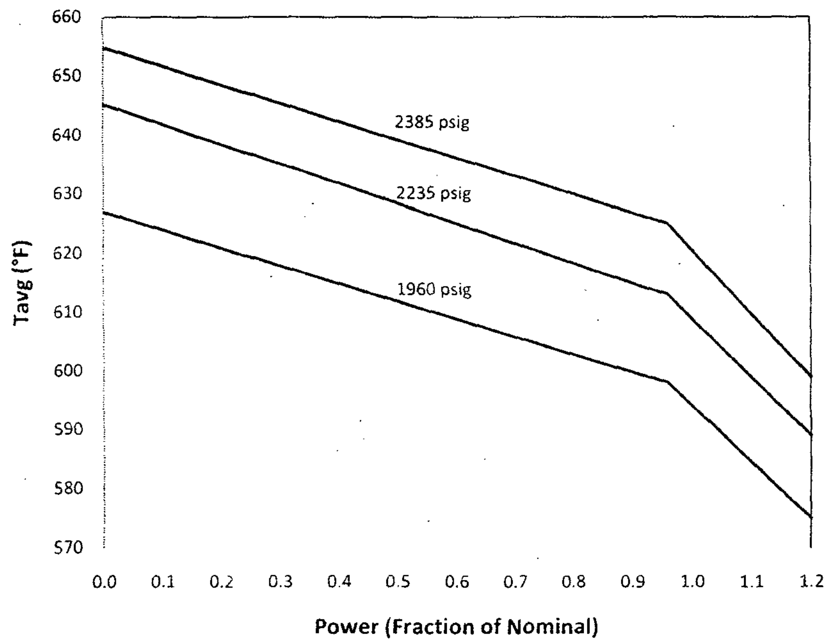

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SEE NOTES 7, 8

INSERT C:

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints." The as-found and as-left tolerances are specified in PLP-106 "Technical Specification Equipment List Program and Core Operating Limits Report."

HNP-12-002

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TAC ME6169

HARRIS NUCLEAR PLANT
SUPPLEMENTAL INFORMATION

Attachment 2
Revised Technical Specification Retyped Pages
(9 pages plus cover)

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

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1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

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1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2948 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

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SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

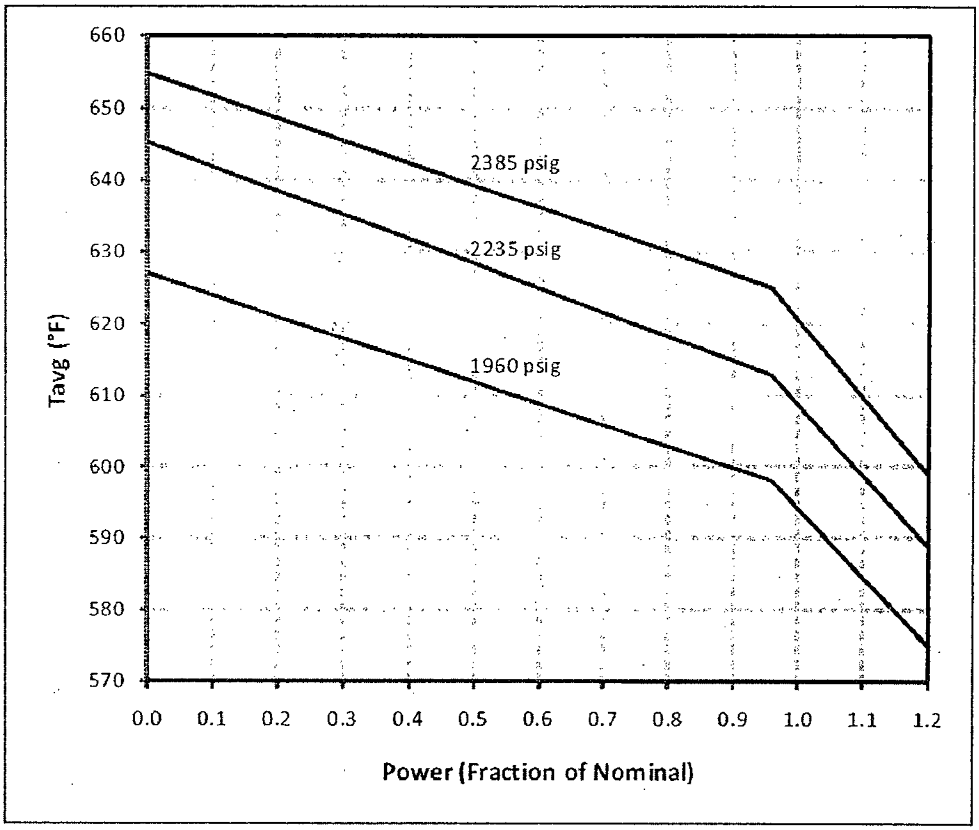

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW ≥ [293,540 GPM X (1.0 + C_1)]

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1.	Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2.	Power Range, Neutron Flux					
	a. High Setpoint	5.83	4.56	0	≤ 108% of RTP** See NOTES 7, 8	≤ 109.5% of RTP**
	b. Low Setpoint	7.83	4.56	0	≤ 25% of RTP** See NOTES 7, 8	≤ 26.8% of RTP**
3.	Power Range, Neutron Flux, High Positive Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5.	Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6.	Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7.	Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8.	Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10.	Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11.	Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥ 1 x 10-10 amp	≥ 6 x 10-11 amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤ 10% of RTP**	≤ 12.1% of RTP**
2) P-13 input	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Inlet Pressure Equivalent	≤ 12.1% RTP** Turbine Inlet Pressure Equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤ 49% of RTP**	≤ 51.1% of RTP**
d. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	≥ 10% of RTP**	≥ 7.9% of RTP**
e. Turbine Inlet Pressure, P-13	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Inlet Pressure Equivalent	≤ 12.1% RTP** Turbine Inlet Pressure Equivalent
20. Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.
22. Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6	=	0.002/°F for T > T" and K_6 = 0 for T≤ T",
T	=	As defined in Note 1,
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤ 588.8°F)
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints." The as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report."

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16.	Underfrequency - - Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6
17.	Turbine Trip (Above P-7)					
	a Low Fluid Oil Pressure	3	2	2	1	6
	b. Turbine Throttle Valve Closure	4	4	1	1	10
18.	Safety Injection Input from ESF	2	1	2	1, 2	13
19.	Reactor Trip System Interlocks					
	a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
	b. Low Power Reactor Trips Block, P-7					
	1) P-10 Input	4	2	3	1	7
	or					
	2) P-13 Input	2	1	2	1	7
	c. Power Range Neutron Flux, P-8	4	2	3	1	7
	d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
	e. Turbine Inlet Pressure, P-13	2	1	2	1	7

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
e. Turbine Inlet Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 9, 10)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M (7, 13) R (14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	49
2	32
3	15

HNP-12-002

Enclosure

SHEARON HARRIS NUCLEAR POWER PLANT / UNIT NO. 1
DOCKET NO. 50-400 / RENEWED LICENSE NO. NPF-63

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE
TAC ME6169

HARRIS NUCLEAR PLANT
SUPPLEMENTAL INFORMATION

Attachment 3
Revised Technical Specification Bases Markups
For Information Only
(9 pages plus cover)

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer, and the RCS piping, pumps, valves and fittings are designed to Section III, Division I of the ASME Code for Nuclear Power Plants, which permits a maximum transient pressure of 110% to 125% of design pressure (2485 psig) depending on component. The Safety Limit of 2735 psig (110% of design pressure) is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. For example, if a bistable has a trip setpoint of 100%, a span of 125%, and a calibration accuracy of 0.5% of span, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is $\leq$ 100.62%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor and an increased rack drift factor, and provides a threshold value for determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Insert 1

~~Manual Reactor Trip~~

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for transport to and response time of the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

inlet

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine ~~impulse chamber~~ pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

inlet

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine ~~impulse chamber~~ pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

inlet

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine ~~impulse chamber~~ pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

Reactor Coolant Flow (Continued)

(a power level of approximately 49% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90.5% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by the setpoint value. The Steam Generator Low Water level portion of the trip is activated when the setpoint value is reached, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients.

inlet

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by the loss of P-7 (a power-level of approximately 10% of RATED THERMAL POWER or a turbine ~~impulse chamber~~ pressure

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine inlet ~~impulse chamber~~ pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, turbine trip, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

Reactor Trip System BASES - Insert 1

<u>Reactor Trip System Instrumentation Setpoints and TSTF-493</u>

This section applies only to the Functional Units to which Notes 7 and 8 in the Trip Setpoint Column are applicable. Those Functional Units have revisions in accordance with Technical Specification Task Force Traveler 493 (TSTF-493), "Clarify Application of Setpoint Methodology for LSSS Functions." Those Functional Units are limited to

- Power Range, Neutron Flux High Setpoint
- Power Range, Neutron Flux Low Setpoint
- Power Range, Neutron Flux High Positive Rate, and
- Power Range, Neutron Flux High Negative Rate

Notes 7 and 8 have been added to Table 2.2-1 that require verifying both trip setpoint setting as-found and as-left values during surveillance testing. In accordance with 10 CFR 50.36, these functions are Limiting Safety System Settings. Adding test requirements ensures that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. These notes address NRC staff concerns with Technical Specification Allowable Values. Specifically, calculated Allowable Values may be non-conservative depending upon the evaluation of instrument performance history, and the as-left requirements of the calibration procedures could have an adverse effect on equipment operability. In addition, using Allowable Values as the limiting setting for assessing instrument channel operability may not be fully in compliance with the intent of 10 CFR 50.36, and the existing surveillance requirements would not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. In the Harris Technical Specifications, the term Trip Setpoint is analogous to Nominal Trip Setpoint (NTSP) in TSTF-493.

Note 7 requires a channel performance evaluation when the as-found setting is outside its as-found tolerance. The performance evaluation verifies that the channel will continue to behave in accordance with safety analysis and instrument performance assumptions in the setpoint methodology. The purpose of this evaluation is to provide confidence in the performance prior to returning the channel to service. If the as-found setting is non-conservative with respect to the Allowable Value, the channel is inoperable. If the as-found setting is conservative with respect to the Allowable Value but is outside the as-found tolerance band, the channel is OPERABLE but degraded. The degraded channel condition will be further evaluated during performance of the surveillance. This evaluation will consist of resetting the channel setpoint to within the as-left tolerances applicable to the actual setpoint implemented in the surveillance procedures (field setting), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to

service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition is entered into the corrective action program for further analysis and trending.

Note 8 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the Trip Setpoint in Table 2.2-1 or within as-left tolerances about a more conservative actual (field) setpoint. As-left channel settings outside the as-left tolerances of PLP-106 and the surveillance procedures cause the channel to be INOPERABLE.

A tolerance is necessary because no device perfectly measures the process. Additionally, it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The as-left tolerance is considered in the setpoint calculation. Failure to set the actual plant trip setpoint to within as-left the tolerances of the NTSP or within as-left tolerances of a more conservative actual field setpoint would invalidate the assumptions in the setpoint calculation, because any subsequent instrument drift would not start from the expected as-left setpoint. The determination will consider whether the instrument is degraded or is capable of being reset and performing its specified safety function. If the channel is determined to be functioning as required (i.e., the channel can be adjusted to within the as-left tolerance and is determined to be functioning normally based on the determination performed prior to returning the channel to service), then the channel is OPERABLE and can be restored to service. If the as-left instrument setting cannot be returned to a setting within the prescribed as-left tolerance band, the instrument would be declared inoperable.

The methodologies for calculating the as-found tolerances and as-left tolerances about the Trip Setpoint or more conservative actual field setpoint are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints," which is incorporated by reference into the FSAR. The actual field setpoint and the associated as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report," the applicable section of which is incorporated by reference into the FSAR.

Limiting Trip Setpoint (LTSP) is generic terminology for the setpoint value calculated by means of the setpoint methodology documented in EGR-NGGC-0153. HNP uses the plant-specific term Nominal Trip Setpoint (NTSP) in place of the generic term LTSP. The NTSP is the LTSP with margin added, and is always equal to or more conservative than the LTSP. The NTSP may use a setting value that is more conservative than the LTSP, but for Technical Specification compliance with 10 CFR 50.36, the plant-specific setpoint term NTSP is cited in Note 8. The NTSP meets the definition of a Limiting Safety System Setting per 10 CFR 50.36 and is a predetermined setting for a protective channel chosen to ensure that automatic protective actions will prevent exceeding Safety Limits during normal operation and design basis anticipated operational occurrences, and assist the Engineered Safety Features Actuation System in

mitigating the consequences of accidents. The Allowable Value is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual NTSP setting ensures that a Safety Limit is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance band, in accordance with uncertainty assumptions stated in the setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

Field setting is the term used for the actual setpoint implemented in the plant surveillance procedures, where margin has been added to the calculated field setting. The as-found and as-left tolerances apply to the field settings implemented in the surveillance procedures to confirm channel performance. A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the instrument operability must be verified based on the field setting and not the NTSP.

HNP-12-002

Enclosure

SHEARON HARRIS NUCLEAR POWER PLANT / UNIT NO. 1
DOCKET NO. 50-400 / RENEWED LICENSE NO. NPF-63

LICENSE AMENDMENT REQUEST
MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE
TAC ME6169

HARRIS NUCLEAR PLANT
SUPPLEMENTAL INFORMATION

Attachment 4
Supporting Information Regarding Terminology Change from
Turbine Impulse Chamber Pressure to Turbine Inlet Pressure
(3 pages plus cover)

Supporting Information Regarding Terminology Change from
Turbine Impulse Chamber Pressure to Turbine Inlet Pressure

Evaluation of the Proposed Changes

1. SUMMARY DESCRIPTION

This letter requests an additional non-technical change to the Renewed Facility Operating License (OL) No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No.1 (HNP) including Appendix A, Technical Specifications (TS), to revise terminology of the function supporting interlock P-13 from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure. As a result of facility modifications supporting operation at uprated conditions, the main turbine will be replaced in the upcoming Refueling Outage. The term Turbine Impulse Chamber Pressure specifically applies to the pressure sensing location on the currently installed turbine. Because the new turbine design does not have an impulse chamber, we request the administrative revision to the functionally equivalent verbiage.

Attachment 1 provides the OL and TS change mark-ups. Attachment 2 provides the retyped OL and TS pages. Attachment 3 provides the TS Bases change (for information only).

2. DETAILED DESCRIPTION

CP&L proposes to revise Renewed Operating NPF-63 pursuant to 10 CFR 50.90.

Revise the terminology from Turbine Impulse Pressure or Turbine Impulse Chamber Pressure to Turbine Inlet Pressure in

TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints, Functional Unit 19. Reactor Trip System Interlocks, b. Low Power Reactor Trips Block, P-7, 2) P-13 input Trip Setpoint, and Allowable Value.

TS Table 2.2-1: Reactor Trip System Instrumentation Trip Setpoints, Functional Unit 19. Reactor Trip System Interlocks, e. Functional Unit description, Trip Setpoint, and Allowable Value

TS Table 3.3-1: Reactor Trip System Instrumentation, Functional Unit 19, Reactor Trip System Interlocks, e. Functional Unit description

TS Table 4.3-1: Reactor Trip System Instrumentation Surveillance Requirements Functional Unit 19, Reactor Trip System Interlocks, e. Functional Unit description.

3. TECHNICAL EVALUATION

There is no technical impact of the terminology change from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure. The existing terminology is a specific sensing location for the function of Turbine Inlet Pressure. The turbine to be installed does not have an Impulse Chamber and the monitoring location will be changed to a functionally equivalent sensing location. Both the existing Turbine Impulse Chamber Pressure and the new Turbine Inlet Pressure measure the inlet pressure to the first full-arc stage which is used as an indirect reactor power equivalent. The proposed changes to replace the phrases "Turbine Impulse Pressure" and "Turbine Impulse Chamber Pressure" with "Turbine Inlet Pressure" does not involve any physical or design change to the P-13 function.

The relationship between turbine inlet pressure and the Rated Thermal Power (RTP) at the new location will be verified during testing. Although the pressure sensed at the new location is higher than the pressure sensed at the current location, the end use devices (i.e., various indication, recording, monitoring, control, and protection functions) of the Reactor Trip System and associated functions will be recalibrated/re-scaled as necessary to maintain their basic functions. The response of the P-13 logic is unaffected, and the design function of the instrument loops has not changed.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The terminology change from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure has no technical impact on existing compliance with applicable General Design Criteria or any other element of the Current Licensing Bases.

4.2 Precedent

This change is similar to changes approved for Beaver Valley Power Station and the Salem Nuclear Generating Station, approved by the NRC on February 24, 2003 (ADAMS Accession No ML030560174) and October 1, 2003 (ADAMS Accession No ML0323701119), respectively.

4.3 Significant Hazards Consideration

The terminology change from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure has no technical impact. Therefore, there is no impact on the Significant Hazards Consideration provided in the original submittal.

4.4 Conclusions

The terminology change from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure has no technical impact. Therefore, there is no impact on the Conclusions of the Significant Hazards Consideration provided in the original submittal.

5. ENVIRONMENTAL CONSIDERATION

The terminology change from Turbine Impulse Chamber Pressure to Turbine Inlet Pressure has no technical impact. Therefore, there is no impact on the Environmental Consideration provided in the original submittal.

6. REFERENCES

a. Beaver Valley Power Station license amendment request, dated August 7, 2002, approved by the NRC on February 24, 2003 (ADAMS Accession No ML030560174)

b. Salem Nuclear Generating Station license amendment request, dated April 10, 2003, approved by the NRC on October 1, 2003 (ADAMS Accession No ML032370119)