CP&L

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U. S. Nuclear Regulatory Commission
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BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
SUPPLEMENTAL INFORMATION TO REQUEST FOR LICENSE AMENDMENTS - POWER UPRATE

Gentlemen:

On July 30 and July 31, 1996, the NRC staff forwarded additional questions to Carolina Power & Light Company (CP&L) on the proposed amendment to the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, which was submitted on April 2, 1996. These proposed amendments revise the BSEP Technical Specifications to allow uprate of the units to 105% of rated thermal power.

Enclosure 1 provides CP&L's formal response to questions raised by the NRC staff on July 30 and July 31, 1996. Enclosure 2 provides a list of regulatory commitments.

Please refer any questions regarding this submittal to Mr. Tony Harris at (910) 457-3312.

Sincerely,

William R Campbell

William R. Campbell

VL/kah

Enclosures

A001 1/1

William R. Campbell, 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 officers, employees, and agents of Carolina Power & Light Company.

Gwendolyn R. Midgette
Notary (Seal)

My commission expires: August 12, 1996

pc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, Jr., NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR LICENSE AMENDMENTS
POWER UPRATE

CP&L RESPONSE TO NRC STAFF QUESTIONS FROM JULY 30, AND JULY 31, 1996

HUMAN FACTORS BRANCH

Question 1

(Section 10.7) Address whether the power uprate will have any effect on operator reliability and performance. Provide examples of operator actions potentially sensitive to power uprate. Other submittals related to power uprate have addressed this question with respect to operator action times assumed in the Individual Plant Examination (IPE) Human Reliability Analysis.

CP&L Response to Question 1

The operator response times identified in GE topical report NEDC-31984P, Supplement 2, were reviewed with respect to the Brunswick Plant PSA model. The GE topical report evaluated the impact of power uprate on a generic basis. The report indicates that some operator response times would be reduced after a power uprate was implemented; however, the effects on operator reliability and performance due to the response time changes were determined to be insignificant.

The Brunswick plant-specific review confirmed that the impact on operator response time is consistent with the generic findings in the topical report. Two operator actions in the Brunswick PSA model require operator responses in a time frame which could be affected by the reduced response times associated with a power uprate. These actions are: 1) manually depressurizing the reactor following a loss of high pressure injection, and 2) injecting boron through the Standby Liquid Control (SLC) system during an ATWS. For both actions, the available operator response time would be reduced by approximately 5%.

The operators must manually depressurize the reactor within about 30 minutes after the initiation of a scram signal if there is a loss of all high pressure injection by manually initiating the Automatic Depressurization System from the control room. The reduction in available operator response time (i.e., approximately 90 seconds) is considered to have an insignificant effect on operator reliability and performance with respect to this action.

Following an ATWS event, the operators must initiate SLC injection within about 10 minutes by actuating control switches on the control board. Simulator observations have shown that operators will typically actuate SLC system in two to three minutes in response to this event. The reduction in available operator response time (i.e., approximately 30 seconds) is within the uncertainty bounds of the human error probability estimates and is considered to be insignificant with respect to this event. Therefore, the operator response time reductions expected as a result of power uprate are considered to have an insignificant effect on operator

reliability and performance with respect to the actions assumed in the Brunswick PSA.

Question 2

(Section 11.1) Discuss any changes the power uprate will have on control room instruments, alarms, and displays. Are zone markings on meters changed (e.g., normal range, marginal range, and out-of-tolerance range)?

CP&L Response to Question 2

As a result of power uprate, there are six (6) control room indicators which will require a face plate scale change. Four (4) of the indicators are for the Main Steam Line Flow indication, the other two (2) are the RWCU system inlet and return flow indicators. In addition to these scale changes, there are two (2) indicators which will require a change to the zone markings. The HPCI and RCIC speed indicators have a normal operating band to 4000 RPM and 4500 RPM, respectively. Due to the change in the speed required for the turbines in both systems, the normal operating band will be increased to 4100 RPM for HPCI and 4600 RPM for RCIC. These increases do not require a faceplate change to the indicators, but only shift the color band that shows the normal range on the instrument. There are no changes necessary for the marginal and out of tolerance ranges.

There are various instructional aids in the main control room that will be revised due to power uprate. These instructional aids are labels, sketches, or markings, which are posted and used as memory or instructional guidance. The list of instructional aids affected by power uprate include the power/flow map, Emergency Operating Procedure (EOP) "Hard Cards" (memory aids), Reactor Pressure vs. Power Table, MSR Reheater 2nd Stage Reheat Tube Bundle Temperature Guidelines, Percent Rated Power vs. Core ΔP (psid), and SRV Setpoints/Locations. Operator impact of power uprate will be communicated through training.

Question 3

(Section 11.1) Discuss any changes the power uprate will have on the Safety Parameter Display System.

CP&L Response to Question 3

The changes to the Safety Parameter Display System (SPDS) include recalibration of Input/Output points, changes to constants which feed composed points (e.g. Rated Thermal Power), changes to the SRV lift setpoints, and changes to the EOP Limit graphs.

The design and intent of the SPDS remains unchanged. These changes will be transparent to the operator in that EOP execution will not be affected.

I&C BRANCH

Question 1

On page E1-8 of the License Amendment Request, item 25 Reactor Vessel - High; Expand on the basis for this setpoint change.

CP&L Response to Question 1

The purpose of the Reactor Vessel - High trip is to shut down the RCIC system by closing the steam supply valve. The RCIC system trip on high reactor vessel water level is assumed in the Loss of Feedwater event analysis (UFSAR Section 6.3.2.8). The Reactor Vessel - High trip shuts down the RCIC system by closing the steam supply valve to prevent overflow of coolant into the main steam lines (MSLs), thereby preventing damage to equipment downstream of the RPV.

As indicated in the License Amendment Request, the analytical limit for this function is not changing as a result of power uprate. The allowable value and trip setpoint have been calculated in accordance with CP&L's setpoint methodology. The resultant trip setpoint of +206.8" is high enough to preclude isolating the injection valve of the RCIC during normal operation, yet low enough to trip the RCIC system prior to water overflowing into the MSLs at approximately +250".

Question 2

On page E1-8 of the License Amendment Request, Item 27 Safety Relief Valves; Provide additional justification for the revision to the Action Statements.

CP&L Response to Question 2

The current Technical Specification Action Statement reflects the analysis performed (ATWS and Vessel Overpressurization) for the Brunswick units which allows 2 SRVs out of service. The change requested in the License Amendment Request was rewritten to adopt the philosophy of NUREG-1433, which details the required number of operable valves in the Limiting Condition for Operation (based on analysis) vice the total number of valves. For power uprate, the most limiting event is the ATWS event, which assumes one (1) SRV out of service. Based on this approach, the Limiting Condition for Operation was revised to indicate 10 of the 11 SRVs are required to be operable, and the Action Statement was revised to require the plant to proceed to HOT SHUTDOWN with less than the required number of valves operable. The proposed revision to the LCO and associated Action Statements does not include consideration for single failure since the single failure is assumed to occur during initiation of the event (i.e., failure to scram or failure to scram from valve position).

MECHANICAL BRANCH

Question 1

In Section 3.3.2 of Reference 1, it is stated that the specific applicable Code Edition for the reactor pressure vessel, including the shroud support, is the 1965 Edition with Addenda to and including Summer 1967." However, the ASME Code, 1971 Edition with Addenda through Winter 1972 is used for the power uprate fatigue evaluation for the feedwater nozzle and core spray nozzle. Please clarify which is the Code used for the power uprate reactor vessel evaluation.

CP&L Response to Question 1

The Code used for the power uprate reactor vessel evaluation was the 1965 Edition with Addenda to and including Summer 1967, which is the Code of construction. However, if a component underwent a design modification, the Code used for the power uprate evaluation of that component was the Code used in the stress analysis of the modified component. The governing Code editions for the modified components are:

Feedwater Nozzle Unit 1: 1971 Edition with Addenda through Summer 1973
Core Spray Nozzle: 1986 Edition, no Addenda
Recirculation Inlet Nozzle: 1986 Edition, no Addenda

In addition, the code edition for the elastic/plastic evaluation of the unmodified U2 Feedwater Nozzle was 1971 Edition with Addenda through Winter 1972.

Question 2

In Section 3.3.2 of Reference 1, provide a discussion on the methodologies and assumptions used in the evaluation regarding the effects of the power uprate on LOCA flow-induced loads. Also, provide methodology and the Code and Edition used for power uprate evaluation and allowable stress for the shroud, shroud support and other reactor internal components. Clarify whether stresses on the shroud support include seismic loads in conjunction with RIPDs and LOCA loads.

CP&L Response to Question 2

1) The LOCA flow-induced loads consist of the short duration flow-induced and acoustic impulse loads resulting from a postulated break of the recirculation suction line.

The flow-induced loads are calculated with the "TRACG" (Transient Reactor Analysis Code-GE version) computer code for three-dimensional finite element flow analysis, using the shroud annulus geometry with power uprate thermal and hydraulic conditions. These loads are highly dependent on the downcomer subcooling conditions; therefore, a LOCA is evaluated over the range of the power/flow map to determine the worst case. The resulting worse case occurred at the minimum MELLA (Maximum Extended Load Line Limit Analysis) point (58% power/33% flow).

Acoustic shock loads are loads impacting components near the recirculation outlet nozzle resulting from the decompression shock wave which originates from a sudden recirculation line break. Acoustic loads are also calculated from the TRACG computer code. A bounding TRACG evaluation was performed using geometry for a larger (BWR-251) size plant, then conservatively applied to the Brunswick structural evaluations.

2) The ASME Code was not specifically applicable to reactor internals during the original design and construction period of Brunswick, therefore there was no specific Code Edition in effect at that time. However, GE used the ASME Code in conjunction with engineering judgement as a basis to determine allowable stresses as specified by the plant UFSAR. For example, Code Sm values are used with appropriate safety factors specified by the UFSAR for upset, emergency, and faulted conditions.

3) The stresses on the shroud support include seismic loads as well as RIPDs and LOCA loads.

CONTAINMENT BRANCH

Question 1

In the GE submittal, NEDC-32466P item 4.1.2.2, it is indicated that SRV flow rates used in the Plant Unique Analysis Report are conservative relative to values calculated with the current analytical setpoints and certified values. Provide example values.

CP&L Response to Question 1

SRV flow rates used in the Plant Unique Analysis Report are conservative relative to values calculated with the current analytical setpoints and certified values as indicated in the following table:

SRV FLOW RATES (all values in lbm/sec)		
SRV Group	PUAR SRV Flow Rate	Current Calculated SRV Flow Rate
I	288.6	277.0
II	291.2	279.6
III	293.8	281.0

The highest calculated power uprate SRV flow rates are as indicated in the following table. Power uprate SRV flow rates include consideration of the setpoint tolerance change from 1% to 3%.:

SRV Group	Highest Calculated Power Uprate SRV Flow Rate (lbm/sec)
I	290.1
II	292.5
III	295.3

Since issuance of the NEDC-32466P, the SRV load evaluation has been reevaluated based on the SRV throat diameters that were assumed in that evaluation. The SRV load evaluation results discussed in the topical report assumed an SRV throat diameter of 4.905". Some of the BNP SRVs have a larger throat diameter of 5.03". The topical report noted that the SRV load increases from power uprate are only 0.1% above the PUAR values; when the larger throat diameter is included in the calculation the load increases to 0.5%. The conclusions in NEDC-32644P that power uprate does not impact the load evaluation for first actuations of SRVs is not changed by this increase.

Question 2

In the above report, item 4.5.1, it is indicated that following a LOCA, the CAD system will be initiated earlier, and the limiting pressure of one-half of the design pressure will be reached earlier. Provide values of the CAD initiating time and limiting pressure.

CP&L Response to Question 2

The CAD initiation referred to in the BNP Power Uprate licensing topical report is a manual initiation. Based on the 5% increase in core power level which would result from power uprate, the required initiation time for the CAD system is expected to be slightly reduced; however, the Brunswick units remain bounded by the generic evaluations performed in NEDO-22155, Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments, June 1982, which has been submitted (transmitted to NRC staff by letters dated June 21, 1982 and August 12, 1982) and reviewed and approved by the NRC staff (see NRC Generic Letter 84-09, dated May 8, 1984). This document forms the basis for CP&L's responses to the issues addressed in NRC Generic Letter 84-09, which were accepted by the NRC staff by letter dated August 20, 1987.

NEDO-22155 indicates that for the case where the initial containment oxygen concentration is 4% (i.e., the maximum allowed by Brunswick Plant Technical Specifications), the combustibility limit of 5% is reached in greater than 1000 days. Conservatisms in this evaluation include 1) the core power to drywell volume ratios assumed (bounding case in NEDO-22155: 0.0301), 2) no containment leakage is assumed to occur, 3) the evaluation was performed assuming initial operation at 102% of core thermal power (bounding case in NEDO-22155: 3359 MWt), and 4) the use of RG 1.7 assumptions for the release fractions of fission products to the reactor coolant and suppression pool. Since the determination of radiolytic oxygen concentration in the containment is based on the ratio of core thermal power to containment free volume, and under power uprate conditions, this ratio at Brunswick is 0.0157, 52% of the bounding plant, the

conclusions of this report (i.e., that the existing inerted BWR Mark I containment design is adequate for control of combustible gases without the need for hydrogen recombiners or containment venting) remain valid for the Brunswick Plant following power uprate.

Since the CAD system initiation is the basis for the limiting pressure, the slight reduction in the required CAD initiation time indicated above would mean that the containment limiting pressure would be reached slightly earlier as a result of power uprate. However, the bounding evaluations of NEDO-22155 still apply. Therefore, the containment limiting pressure of 31 psig would not be expected to be reached until greater than 1000 days following the event (following CAD initiation).

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
NRC DOCKET NOS. 50-325 & 50-324
OPERATING LICENSE NOS. DPR-71 & DPR-62

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not considered regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed Date
No New Commitments Are Contained In This Letter	N/A