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Vice President
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TSC-2001-09

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


BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING
REQUEST FOR LICENSE AMENDMENTS - EXTENDED POWER UPRATE
(NRC TAC NOS. MB2700 AND MB2701)

Ladies and Gentlemen:

On August 9, 2001 (i.e., Serial: BSEP 01-0086), Carolina Power & Light (CP&L) Company requested a revision to the Operating Licenses (OLs) and the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments increase the maximum power level authorized by Section 2.C.(1) of OLs DPR-71 and DPR-62 from 2558 megawatts thermal (MWt) to 2923 MWt. Subsequently, on February 21, 2002 (i.e., Serial: BSEP 02-0038), CP&L responded to a Request for Additional Information (RAI) concerning the difference in turbine bypass valve capability of the two BSEP units and how this difference was addressed in the BSEP probabilistic safety analysis of the planned extended power uprate. On February 25, 2002, the NRC provided an electronic version of a follow-up RAI concerning turbine bypass valve capability. The response to this follow-up RAI is enclosed.

Please refer any questions regarding this submittal to Mr. David C. DiCello,
Manager - Regulatory Affairs, at (910) 457-2235.

Sincerely,


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A001

MAT/mat

Enclosure:

Response to Request for Additional Information (RAI) 22

C. J. Gannon, 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.

Dean S. Mason
Notary (Seal)

My commission expires: August 29, 2004

cc:

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ENCLOSURE

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Response to Request for Additional Information (RAI) 22

Background

On August 9, 2001 (i.e., Serial: BSEP 01-0086), Carolina Power & Light (CP&L) Company requested a revision to the Operating Licenses (OLs) and the Technical Specifications for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments increase the maximum power level authorized by Section 2.C.(1) of OLs DPR-71 and DPR-62 from 2558 megawatts thermal (MWt) to 2923 MWt. Subsequently, on February 21, 2002 (i.e., Serial: BSEP 02-0038), CP&L responded to a RAI concerning the difference in turbine bypass valve capability of the two BSEP units and how this difference was addressed in the BSEP probabilistic safety analysis of the planned extended power uprate (EPU). On February 25, 2002, the NRC provided an electronic version of a follow-up RAI concerning turbine bypass valve capability. The response to this RAI follows.

NRC Question 22-1

On page 4-48 of Enclosure 2 of the November 30, 2001 response to my RAIs (Serial: BSEP 01-0141), it states that for an ATWS without SLC injection, the power level reached would be 21.6%. This is greater than the Unit 1 TBV capacity, which would mean that for this scenario the pathway cannot be considered a success (i.e., some steam must be dumped to the suppression pool via the SRVs).

Based on the TBVs not being adequate for the above scenario, I again request that the licensee determine the risk implications (i.e., calculate the delta CDF and delta LERF) of not being able to consider the condenser (via the TBVs) as a valid pathway for dumping the steam for this scenario. This could be done as a sensitivity case to the base case. However, if it indicates the potential for a large change in risk, a combined sensitivity case would need to be also run that incorporates the other sensitivity cases with this one.

Response to Question 22-1

As discussed in the response to RAI 15-1 (i.e., Serial: BSEP 02-0038) EPU will reduce the turbine bypass capacity from 23.79% to 20.6% for Unit 1 and from 80.26% to 69.6% for Unit 2.

Section 7.1.3 of the PUSAR concludes that this capacity is adequate for EPU operation. Specifically, Section 7.3 states:

The steam bypass system is a nonsafety-related system. Even though the bypass capacity as a function of the percent uprated steam flow is reduced, the actual steam bypass capacity is unchanged. This capacity is used in the transient analysis (Section 9.1) for the evaluation of events that credit the turbine bypass system availability. Because the EPU transient analysis results are acceptable, the turbine bypass capacity is adequate for EPU operation.

Therefore, the steam bypass system remains able to perform its intended function under EPU conditions.

In response to an actual Anticipated Transient Without Scram (ATWS) event, BSEP Emergency Operating Procedures (EOPs) would direct operators to reduce power, as quickly as possible, to less than 4% rated thermal power. Several strategies, such as boron injection via the Standby Liquid Control (SLC) system, tripping of the recirculating pumps, and reactor water level control; coincident with continued attempts to insert control rods are employed. These efforts will be continued until the reactor is shutdown.

The power level estimated on page 4-48 of Enclosure 2 of CP&L's November 30, 2001, submittal (i.e., Serial: BSEP 01-0141) is an extrapolation of Probabilistic Safety Analysis (PSA) analyses and should not be directly compared with the values calculated in the PUSAR. The values are calculated using different codes, assumptions and boundary conditions. The discussion is part of a section of the PSA evaluation that examined possible EPU effects from the perspective of accident sequence progression. The main conclusion is that the increased power level reduces the time available to perform operator actions.

Additionally, there are no risk implications even if it is assumed that Unit 1 reactor power will exceed turbine bypass capability following an ATWS. The BSEP Probabilistic Safety Analysis (PSA) model defines as core damage, any ATWS scenario combined with the failure to lift any 6 of 11 Safety Relief Valves (SRVs). This implies that any ATWS scenario combined with failure to lift 6 SRVs yields minimal core damage sequences. As such, quantification of any ATWS sequence, combined with the failure to lift any 6 of 11 SRVs, and any other additional failures yields non-minimal cutsets that do not contribute to the Core Damage Frequency (CDF). After EPU, the SRVs may temporarily lift for a longer duration but the Turbine Bypass Valve (TBV) and SRV success criteria would not change based on the relative reduction in bypass capability associated with EPU. Furthermore, if it is postulated that an additional SRV is required to lift during ATWS scenarios, the impact to risk would remain negligible since the failure to lift sufficient SRVs is dominated by common cause failure.

It should be noted that, at the onset of an ATWS scenario (i.e., T_0), under both current and EPU conditions, the initial reactor power would most likely exceed the TBV capacity, but the power level would be suppressed quickly to a point where the bypass capacity is adequate. Failure to quickly reduce the power during an ATWS scenario is defined, in the BSEP PSA model, as core

damage. This is modeled in the PSA as ATWS sequences combined with failure to trip any recirculation pump, or failure to lift an adequate number of SRVs, or failure of SLC system injection. Since the success criteria between the pre-EPU and post-EPU remains the same for these sequences in the BSEP PSA analyses, there is no change to CDF or Large Early Release Frequency (LERF).

Based on the above discussion, it is concluded that the relative changes in turbine bypass capability are small and do not impact the success criteria in the BSEP Unit 1 PSA analysis.