CP&L

Carolina Power & Light Company

JUL 01 1982

Office of Nuclear Reactor Regulation
ATTN: Mr. D. B. Vassallo, Chief
Operating Reactors Branch No. 2
United States Nuclear Regulatory Commission
Washington, D.C. 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
RELOAD LICENSING - ADDITIONAL INFORMATION

Dear Mr. Vassallo:

In accordance with our June 14-15, 1982 meeting with NRC staff members, Carolina Power & Light Company (CP&L) hereby submits the Emergency Core Cooling System (ECCS) analysis to support information provided to the staff in that meeting. The remaining structural and loose parts analyses will be submitted to you in the near future.

This ECCS analysis was preceded by revised Technical Specifications (TS), dated June 28, 1982, which incorporated a uniform 8.5 percent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) reduction imposed pursuant to the ECCS information presented in the aforementioned meeting.

Brunswick Unit No. 2 criticality is presently scheduled for July 10, 1982. We request the Staff review be conducted to support issuance of the TS revisions to allow return to power in accordance with this schedule. We will keep your staff apprised of any changes to the schedule.

Should you require additional information, please contact our staff.

Yours very truly,

S. R. Zimmerman
Manager
Licensing & Permits

SRZ/ce (205C4T4)
Enclosure

cc: Messrs. J. P. O'Reilly (NRC-RII)
J. A. Van Vliet (NRC)

A001

411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

LOSS-OF-COOLANT ACCIDENT ANALYSIS WITH A CRACK IN ONE CORE SPRAY SPARGER

1.0 Introduction

A crack in the core spray sparger at the Brunswick 2 plant, located in the upper sparger near the T-Box to sparger weld, has been clamped. The structural integrity of the sparger and the intended cooling function of the spray system are not adversely affected by the presence of the crack. Therefore, no change in Emergency Core Cooling System (ECCS) performance analysis or Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits is required. However, at the request of Carolina Power & Light Company (CP&L), conservative ECCS sensitivity analyses have been performed and are presented herein.

This section describes the methods used to evaluate the MAPLHGR requirements to meet 10CFR50 Appendix K for the current Brunswick 2 operating cycle, with a cracked core spray sparger. The potential effect of spray distribution on the Peak Cladding Temperature (PCT) of the limiting break size and a single failure is discussed in Section 2. The phenomena involved and the inputs to the approved 10CFR50 Appendix K computer codes are discussed in Section 3, the results of analyses performed are given in Sections 4 and 5, and the conclusions presented in Section 6.

2.0 Limiting Break Size and Single Failure Analysis

For the Brunswick 2 plant, there are no postulated single failures for any break location (other than a core spray line break) that can result in less than one core spray system injecting water into the upper plenum above the reactor core. For a core spray line break, there are always at least three low pressure ECCS pumps injecting water into the reactor vessel, thereby assuring that the break is not a limiting event. For medium and large break sizes (which depressurize relatively quickly), the most limiting failures are those that result in the least number of ECCS pumps remaining operable (i.e., injecting water into the reactor vessel).

The only two single failure candidates that are potentially limiting for medium to large break sizes are:

A. Diesel Generator Failure - 1 core spray (LPCS) + 1 Low Pressure Coolant Injection (LPCI) + HPCI + the ADS operable;

B. LPCI Injection Valve Failure - 2 core spray (LPCS) + HPCI + the ADS operable.

Since the High Pressure Coolant Injection (HPCI) is steam turbine powered, it is not a significant contributor to mitigating medium to large breaks which depressurize rapidly. Also, since the function of the Automatic Depressurization System (ADS) is to depressurize the reactor as a backup to the HPCI, it contributes little toward mitigating medium and large break LOCAs. Therefore, failure candidates A and B are limiting and each result in a dependence on only two ECCS pumps.

Per the Brunswick 2 (Reload 4, Cycle 5) reload analysis, failure candidate B (LPCI Injection valve failure) is limiting because of the conservative modeling of counter current flow limiting (CCFL) at the fuel assembly upper tie plates. The calculation limits the coolant delivery or downflow from the core spray systems to the fuel bundles and further prolongs core reflooding by neglecting the water held back in the upper plenum.

Both single failure candidates (A and B) were re-examined for large breaks acknowledging the crack in one sparger. The limiting single failure, break size, and location were found to not change. This is because the calculated core uncovery and recovery times, and the reactor depressurization rates are insensitive to changes in spray cooling heat transfer, and even with more realistic treatment of CCFL, the failure candidate B yields (marginally) the limiting result.

For smaller break sizes, the limiting single failure is the high pressure ECCS (HPCI) since the transient is a relatively slow depressurization event that is dominated by the time required to either reflood the reactor with the high pressure system or the time to depressurize the reactor so that the low pressure systems become effective. Furthermore, the effects of CCFL in limiting coolant delivery to the core are not as large at high reactor pressures. The small break LOCA transient is, therefore, insensitive to spray distribution because reflooding occurs very rapidly once any one or two of the six low pressure ECCS pumps begin injecting coolant into the reactor vessel.

Therefore, only medium and large break LOCA calculations have any potential for dependence upon spray distribution, and detailed LOCA calculations need only be performed for large limiting break sizes with the current limiting single failure.

3.0 Phenomena Involved in the Analysis of Sparger Performance

The key phenomena involved in evaluating core cooling performance resulting from the injection of spray through the core spray sparger in the BWR are listed in Table 1. The analytical assumptions regarding these phenomena which are important to understanding system performance and the predicted core cooling are also tabulated.

The approved Appendix K models include these phenomena, but the input assumptions used in the reload analysis are overly conservative. The extent of this conservatism is evident from Table 1 in light of the realistic phenomena observed and tabulated. The bases for the first three of these realistic inputs are derived from a recently completed, jointly sponsored, large scale BWR safety research program between NRC, EPRI, and GE (Reference 1).

The relevant phenomena do not depend on the distribution of the injected spray through the nozzles but on the injection of the coolant into the upper plenum (Reference 1). Recently, the NRC staff has evaluated the issues related to the adequacy of the core spray systems in the BWR and their ability to distribute spray water to the core (Reference 2). This evaluation was in response to concerns that the core spray systems may not distribute any spray to certain regions of the core when injected into an upper plenum steam environment. The staff testimony in Reference 2 concluded that the spray distribution adequacy is not a safety concern because the coolant injected into the upper plenum will either disperse uniformly in a pool of water above the core or will flow to the lower plenum producing rapid reflooding. Therefore, the current reload calculation is applicable and conservative despite the presence of any crack(s) in the core spray sparger.

Table 1 Key Phenomena Related to Core Spray Cooling Performance

Phenomena	Analytical Assumptions Used in the Current Reload Analysis	Realistic Assumptions
Upper Plenum Inventory	Conservatively assumed to not interact or contribute to core reflood during LOCA transient	Pool of water present throughout transient assures coolant delivery to all fuel bundles (supported by Large Scale Tests)
Counter Current Flow Limiting	Saturated water in upper plenum above core	Some subcooling and less CCFL occurs. A residual pool of water remains during and after core reflooding. (supported by Large Scale Tests)
	No CCFL breakdown	Breakdown of CCFL shortly after spray initiation causes rapid reflooding (supported by Large Scale Tests)
Core heat transfer	Limited spray cooling after blowdown Appendix K credit only	Steam cooling contribution as much as 10 times greater than Appendix K spray cooling
Decay Heat	1971 ANS + 20% specified by Appendix K	1979 ANS (GE has submitted a technical basis as a part of the Standard Plant docket which is based on the 1979 ANS decay heat correlation)

4.0 Analysis Results

The current reload analysis for the limiting LOCA with the most limiting fuel type and exposure combination results in a calculated PCT of 2200°F. This is for 8x8R fuel at an exposure of 20,000 MWd/t and a MAPLHGR of 11.8 KW/ft.

Figure 1 shows the heat transfer assumed as a function of time (Curve 1) compared with the realistic heat transfer (Curve 2). A bounding calculation (Curve 3) of the limiting LOCA with approved Appendix K models with CCFL breakdown input based on observed large scale tests, and no convective core cooling prior to reflooding results in a maximum PCT of less than 1260°F at a MAPLHGR of 11.8 KW/ft. This result demonstrates that the current reload calculation is conservative by more than 940°F. No credit for steam cooling or the improved decay heat correlation are included in this calculation which would further reduce the PCT.

A comparison of the current reload analysis with the conservatively calculated PCT using CCFL breakdown is shown in Curve 4 at the bottom of Figure 1. It is clear from this figure that the overly conservative treatment of CCFL results in the unrealistically slow core reflooding time and high calculated PCT in the reload analysis.

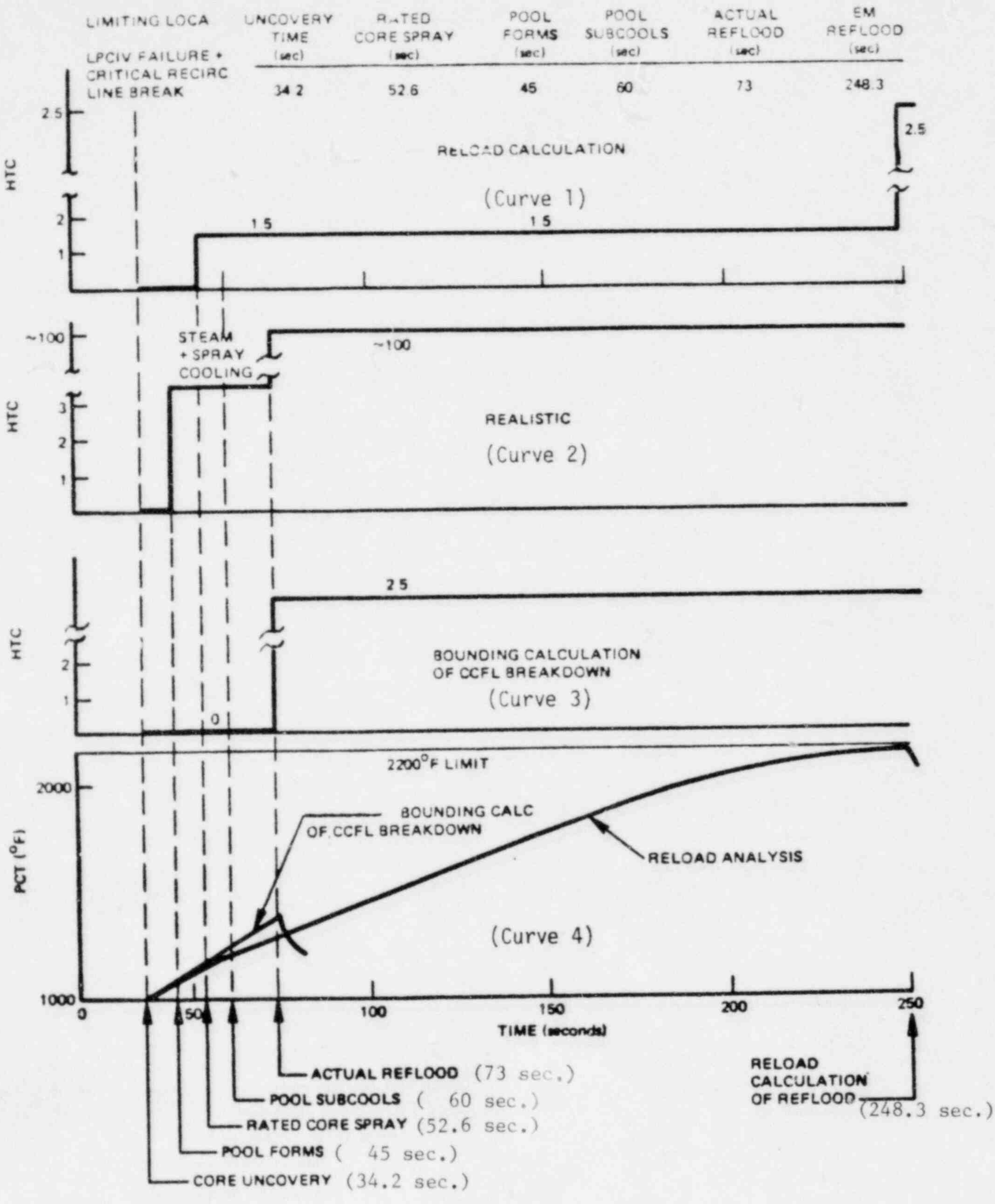

Figure 1. Brunswick 2 DBA (Limiting LOCA) Analysis

5.0 Bounding Sensitivity Calculation

At the request of CP&L, a simplified and conservative sensitivity calculation was also performed with the following assumptions: 1) no credit for upper plenum inventory, 2) no CCFL breakdown, and 3) no cooling contribution from the sparger with the crack before core reflooding. This analysis results in a bounding MAPLHGR reduction of 8.5 percent at 20,000 MWd/t exposure to meet the 2200°F Appendix K limits for all types and exposures.

To perform this bounding calculation, the cooling contribution from the cracked sparger was ignored prior to reflooding. With only the core spray system with the cracked spray sparger operating, the spray heat transfer coefficient was set to zero. For the core spray system with the uncracked sparger operating, the spray heat transfer was set to one half the value used in the reload calculation (Reference 3). This is the approved assumption when one spray system is totally inoperable.

General Electric considers this calculation to be unrealistic and overly conservative but has performed this calculation at the request of CP&L to quantify the sensitivities of the core spray performance to various assumptions.

6.0 Conclusions

An analysis of one cracked core spray sparger in the Brunswick Unit No. 2 BWR was performed utilizing the approved Appendix K evaluation models. The results of this analysis demonstrate that with CCFL breakdown (derived from a conservative interpretation of recent large scale tests) the calculated PCT is at least 940°F less than the current reload calculation. Without CCFL breakdown the upper plenum inventory (pool of water) ensures adequate coolant delivery to the core. Therefore, the current reload calculation is applicable and conservative and there is no basis to impose a MAPLHGR penalty on Brunswick 2 for Cycle 5 or succeeding cycle operation.

7 References

1. Refill-Reflood Program Task 4.4-CCFL/Refill System Effects Test (30°F Sector) SSTF System Response Test Results, D.G. Schumacher, T. Eckert, J.A. Findlay, General Electric Co., NUREG/CR-2568, EPRI NP-1525, GEAP-24893-2, March 1982.

2. Transcript of Testimony before the Atomic Safety and Licensing Board For Long Island Lighting Co., May 18, 1982 - Docket No. 50-322-OL. NRC Staff Testimony of Summer B. Sun on ECCS Core Spray.

3. SER, O.D. Parr (NRC) to G.G. Sherwood (GE), "Review of General Electric Topical Report NEDO-20566, Amendment 3," June 13, 1978.