Florida Power
CORPORATION

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Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
NUREG-0737, Item II.K.3.30
Revised Small Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K.

Dear Mr. Denton:

On July 20, 1982, representatives of the Babcock and Wilcox (B&W) Owners met with your Staff to culminate the continuing dialogue on the scope of the program for resolution of NUREG-0737, "Clarification of TMI Action Plan Requirements", Item II.K.3.30. This letter formalizes the proposals made at that meeting.

We will resolve the two separate areas identified by the Staff in the April 16, 1982 meeting. The first, assurance of core cooling (10CFR50, Appendix K), is being evaluated under the ongoing Small Break Loss of Coolant Accident (SB LOCA) Methods Program with which the Staff concurred. The B&W Owners will continue to address the staff issues of Item II.K.3.30 in the SB LOCA Methods Program as identified in Attachment #1. The B&W Owners have also prepared a number of reports as a result of the recent joint test evaluation with the Staff which are identified in Attachment #2.

The second area deals with the analytical basis for recovery of natural circulation, long term cooling, and operator guidelines and training for these events. The B&W Owners propose to benchmark our best estimate codes with Integral System Test (IST) data from the GERDA SB LOCA test facility. (GERDA stands for Geradrohr Dampferzeuger Anlage, which is German for "straight-tube steam generator test facility"). This facility was designed to provide a better understanding of the long term response of the B&W designed system. It will also provide data which

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will validate Anticipated Transient Operating Guidelines (ATOG) assumptions for these transient periods. The inclusion of GERDA test data should also alleviate the general uneasiness expressed by the Staff in our meetings regarding the need for improved understanding of the B&W design. GERDA will provide test data for natural circulation, interruption of natural circulation, the transition to boiler-condenser mode of cooling, and the long term cooling of the system. This additional data should provide the Staff with sufficient confidence in the validity of B&W's best estimate codes to accept the Owners' program as resolution of Item II.K.3.30.

The B&W Owners are not willing to commit to an open-ended test program but do recognize that issues may be identified as data is analyzed which require further evaluation. We propose to evaluate any issues which arise and to take appropriate action for their resolution.

The following discussion presents more detail in support of this position.

Background

Following the accident at TMI-2, the NRC required that further SB LOCA analyses be performed, and that operator guidelines for managing SB LOCA be developed. The results of this work were documented by B&W on May 7, 1979. In their review, documented in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants", the NRC concluded that while there was not a safety concern, certain features of the B&W SB LOCA Evaluation Model required more extensive verification. In general, the recommendations were:

1. Additional code model predictions of Semiscale and Loss-of-Fluid Test (LOFT) experiments should be performed.

2. The SB LOCA methods should be revised to address the NRC's specific concerns. In addition, the licensees should verify the evaluation models with appropriate integral system data.

These recommendations were implemented as requirements in Item II.K.3.30, NUREG-0737, and the following describes our actions towards resolution of this item.

Discussion

The B&W Owners have taken several actions in responding to these recommendations. In response to Recommendation 1, computer code simulations of LOFT tests L3-1[1] and L3-6[2] and Semiscale test S-07-10D[3] were submitted. The B&W simulation results compared well with the test data and the simulations presented by other vendors.

Since configurations tested in Semiscale and LOFT do not reflect all plant designs and arrangements, the acceptance by the Staff of benchmarks by other vendors would also seem to be applicable to B&W benchmarks of the same tests as adequate testing of computer codes used in SB LOCA calculations.

Prior to any action to respond to the SB LOCA issues in NUREG-0565, the B&W Owners Group met with the Staff on December 16, 1980, to obtain a better quantification of the Staff's concerns relative to NUREG-0737, Item II.K.3.30. The Staff's concerns were specified in the Staff minutes of that meeting.[4]

On May 12, 1981, the Owners Group again met with the Staff to present their program designed to address the issues of Reference 4. The Staff concluded that eight of the nine issues would be resolved by the implementation of the program presented, but that IST data would be required before Item II.K.3.30 could be accepted by the Staff. Attachment #1 details the response to each of the nine items in Reference 4. During the meeting, the Staff raised a number of issues over and above those originally quantified as Item II.K.3.30 issues. Following this meeting and for several months thereafter, a continuing technical dialogue was held between the Owners and the Staff in an effort to obtain and understand a complete list of specific issues.

Finally, in a meeting on October 23, 1981, with B&W Utility Executives, the Staff identified the issues as uncertainties regarding hot leg "bubble dynamics" during the transition from natural circulation to the boiler-condenser mode of cooling. From that meeting, the Staff agreed to participate in an in-depth review of the then current B&W SB LOCA Methods Program, including the verification base. At the same time, the Owners agreed to participate in a joint effort with the Staff to assure that current SB LOCA methods and ATOG programs are fully understood. The program was to include the following:

- Code parameters, models, assumptions, etc., which are important in controlling dynamics of interest will be identified and available experimental data substantiating their validity will be reviewed. This would be done using results of the improved evaluation model in order that the most accurate dynamic response characteristics are reviewed.

- Additional existing experimental data, from separate effects or integral tests, will be identified which addresses specific technical gaps, if any.

- Identify where and how additional experimental data may be obtained, if required.

The Owners Group Analysis Subcommittee set a meeting with the Staff for December 16 and 17, 1981, to implement this commitment. The Owners came to that meeting prepared to address "bubble dynamics" and the CRAFT code. The Staff expected to be presented with a test program and the meeting ended in an impasse. In a letter to the Staff on February 5, 1982, the Subcommittee again set a meeting to discuss:

- phenomena of bubble dynamics
- sensitivity of the system to decay heat, number of high pressure injection pumps, phase slip, and interphase heat transfer
- discussion of benchmarks.

On April 9, 1982, six reports were hand delivered to the Staff for review prior to the April 16, 1982 meeting with the Owners Group. Attachment #2 to this letter provides a brief description of these reports.

In the period between February and April 1982, the Staff again presented issues that were beyond Item II.K.3.30 (Reference 5). Since the Owners were involved in an intensive effort to produce documents in response to the identified focused issue of "bubble dynamics", it was not possible to address the items in Reference 5 specifically in the April 16, 1982 meeting. The presentations in the April 16, 1982 meeting were perceived by the Owners as being well received by the Staff, and to date, no negative comments have been received from the Staff on that meeting. We have since addressed these issues (Attachment #3).

At the conclusion of the April 16, 1982 meeting the issues could clearly be separated into two parts. One part deals with the assurance of core cooling (10CFR50, Appendix K) and the other deals with the analytical basis for recovery of natural circulation, long term cooling, and operator guidelines and training for these events. At this time the Owners began to develop the program described above for acquiring IST data to benchmark best estimate codes to be used in calculating operator oriented phenomena for ATOG.

Summary

The B&W Owners are continuing their work to address Item II.K.3.30 with the SB LOCA Methods Program described to the Staff and with the six reports described in Attachment #2. We further offer to benchmark best estimate codes with GERDA test data to provide better Staff understanding of the concerns in Reference 5 which are beyond Item II.K.3.30. We believe that GERDA is a technically acceptable test facility to address the phenomenon associated with recovery from a SB LOCA and offers a unique way to benchmark several of these phenomenon as they interrelate; that is, GERDA is an integral system test focused on the longer term natural circulation phenomena of the B&W design. We have provided the Staff with technical presentations on the design of GERDA at the Alliance Research Center on July 7, 1982, and followed with a tour of the facility.

The majority of the Staff comments during and immediately following the presentation were favorable. However, a very negative comment as to the applicability of GERDA was made by the Staff in the July 20, 1982, meeting with the B&W Utility Executives. We would be happy to address any technical questions the Staff or their consultants might have regarding GERDA and the test program at the facility. B&W will send you, under separate cover, a description of the GERDA test program.

We view our IST test program as the final element in addressing the issues raised by the Staff during their review of the Item II.K.3.30, SB LOCA Program and as a source of useful data to address other issues. These tests will be used as the bridge in the next logical step towards identifying any residual need for additional or modified test facilities. We therefore invite the Staff to consider our test program as the means to minimize limited Owner and Staff resources while enhancing the knowledge of the B&W system.

We intend to provide a follow-up letter within the next three weeks which will provide additional details and milestones which we intend to pursue.

Very truly yours,

Patsy Y. Baynard

Patsy Y. Baynard
Assistant to the Vice President
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Attachments

Bright(T01)D153-3

cc: Mr. J. P. O'Reilly
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References

1. "B&W's Post Test Evaluation of LOFT Test L3-1", Document No. 51-1125988-00, May 1981.

2. "B&W's Best Estimate Prediction of the LOFT L3-6 Nuclear Small Break Test Using the CRAFT 2 Computer Code", Document No. 12-1124993-01, March 1981.

3. "B&W's Post Test Analysis for Semiscale Test S-07-10D", Document No. 86-1125888-00, May 1981.

4. Summary of Meeting with the B&W Owners Group Concerning the Abnormal Transient Operating Guidelines (ATOG) Program and NUREG-0737, Item II.K.3.30 Small Break Loss of Coolant Accident Models (December 16, 1980).

5. Letter from Eisenhut to Mattimoe, March 25, 1982, Docket No. 50-312, Subject: Need for Model Verification.

ATTACHMENT #1

Nine areas of concern for II.K.3.30 were identified in the meeting of December 16, 1980 between the Staff and B&W Owners. These concerns are repeated below as found in the minutes of that meeting prepared by Mr. Throm of the Reactor Systems Branch. Owner responses to each concern are also included.

1. NEED TO VERIFY THE CURRENT NON-CONDENSIBLE MODEL AND THE CONSERVATISM OF THE CONDENSATION HEAT TRANSFER RATE IN THE STEAM GENERATOR.

   a) Report has been prepared describing a method to predict the amount of non-condensible gases in the primary system, including gas produced via radiolytic decomposition which may be released during a SBLOCA. This report will be submitted to the NRC in August 1982.

   b) A non-condensible gas heat removal model has been prepared and incorporated into the CRAFT code. This model is described in the revision to the CRAFT Topical Report scheduled for submittal to the Staff in September 1982.

2. NEED TO VERIFY THE NON-EQUILIBRIUM MODEL AND TO JUSTIFY THAT THE AMOUNT OF EMERGENCY CORE COOLING SYSTEM (ECCS) WATER INJECTION IS CONSERVATIVE.

   a) Report has been prepared and will be submitted to the Staff in August which justifies the current B&W ECCS evaluation model which utilizes CFT injection into the lower downcomer region.

   b) This work was discussed with the Staff in the technical presentations on December 16, 1981.

3. NEED TO DISCUSS THE PRESSURIZER MODEL AND THE EFFECTS OF A NON-EQUILIBRIUM MODEL.

   a) A non-equilibrium pressurizer model has been incorporated into the CRAFT code. This model will be addressed in the revised CRAFT Topical Report to be submitted to the Staff in September 1982. This model was discussed with the Staff on December 16, 1981.

   b) The surge line model was discussed with the Staff on December 16, 1981. The open question from the Staff will be addressed in a written response in September 1982.

4. NEED TO ADDRESS THE FORMATION OF A STEAM BUBBLE IN THE HOT LEG "CANDY CANE". (IS IT A REAL OR CALCULATED PHENOMENON?) EXPERIMENTAL VERIFICATION BELIEVED NECESSARY.

   a) This is addressed in several parts of the SBLOCA Methods Program:

      . System modeling study (steam generator, hot leg, and reactor vessel head)

      . Steam generator and pressurizer model changes

   b) The joint NRC/Owners testing evaluation task concentrated on this issue. Documents described in Attachment #2 support the evaluation of this concern, and the report on "Bubble Dynamics" specifically addresses this concern.

5. THE STAFF INDICATED THAT A MECHANISTIC MODEL OF THE STEAM GENERATOR HEAT TRANSFER SHOULD BE DEVELOPED. A BEST ESTIMATE OR VERIFIED CONSERVATIVE MODEL WOULD BE ACCEPTABLE.

   a) The steam generator model has been upgraded and will be described in the revision of the CRAFT Topical Report to be issued to the Staff in September 1982.

   b) The steam generator model was presented to the Staff in the December 16, 1981 meeting.

6. AS PART OF THE ADDITIONAL SYSTEMS VERIFICATION NEEDED, THE FOLLOWING SEMISCALE AND LOFT TESTS SHOULD BE CONSIDERED: SEMISCALE S-07-10D, LOFT L3-1, L3-5, and L3-6.

   a) The Owners considered the above tests and provided the Staff post test evaluations of L3-1, L3-6, and S-07-10D (References 1, 2, and 3 to this letter).

7. THE OVERALL THERMAL-HYDRAULIC BEHAVIOR OF THE CORE DURING UNCOVERY SHOULD BE VERIFIED AGAINST APPLICABLE EXPERIMENTAL DATA, PARTICULARLY THE RECENT OAK RIDGE NATIONAL LABORATORY (ORNL) DATA.

   A) ORNL data has been used to show that the current application of the Ditters-Boelter correlation is conservative. Data was discussed with the Staff on December 16, 1981, and a report will be provided to the Staff in August 1982.

8. THE INFLUENCE OF METAL HEAT ON THE SYSTEM PRESSURE RESPONSE, PARTICULARLY ON THE TIME OF ECCS INJECTION, WAS IDENTIFIED AS AN AREA OF CONCERN AND SHOULD BE SHOWN TO BE PROPERLY CONSIDERED IN THE ANALYSIS MODELS.

   a) The B&W ECCS Evaluation Model currently accounts for metal heat and no change needs to be made.

9. THE BREAK FLOW MODEL NEEDS TO BE CONFIRMED. THE USE OF COMBINED MODELS WITH VARIOUS DISCHARGE COEFFICIENTS APPLIED TO THEM NEEDS TO BE COMPARED TO A BEST ESTIMATE MODEL TO DEMONSTRATE CONSERVATISMS.

   a) The existing leak discharge model has been found to produce results which are similar to yet still conservative with respect to those obtained with the best estimate model.

   b) The work was discussed with the Staff on December 16, 1981 and the report will be provided to the Staff in August 1982.

ATTACHMENT #2

Documents prepared and submitted to the Staff from the B&W Owner's participation in the joint test evaluation task with the NRC.

"The GERDA Test Facility"

This report was prepared in fulfillment of the October 23 commitment by B&W.

"CRAFT 2 Prediction of Alliance Research Center (ARC) Loss-of-Feedwater Test", 12-1132544-00, April 1982

This report shows that the revised steam generator model adequately predicts the temporal response of key once-through steam generator (OTSG) parameters after a complete loss of feedwater.

"Auxiliary Feedwater Penetration", 12-1132513-00, April 1982
"Auxiliary Feedwater Axial Flow Distribution", 12-1132543-00, April 1982

The first report describes the calculation model and testing basis for the penetration of the auxiliary emergency feedwater AFW in the OTSG, and the second report uses this model and shows how the axial flow distribution was derived from first-of-a-kind testing at Oconee 1.

"Benchmarks for AFW Models", 12-1132555-00, April 1982

This report contains the benchmark results of the AFW models against actual plant data from four plant transients. The ability to predict plant response following loss of offsite power for the extreme conditions under which the AFW system will function is demonstrated in this report.

"Bubble Dynamics", 12-1132565-00, April 1982

This report is focused on the main phenomenological aspects of steam in the hot leg "U" bend and addresses test data and engineering evaluation used to understand "bubble dynamics". Based upon the focused Staff concern on the dynamics of a trapped steam bubble in the inverted U-bend of the hot legs, two issues were identified:

1. During the blowdown portion of the transient, does the code properly predict the formation of the steam bubble and its resultant interruption in natural circulation?

2. During the system refill phase of the transient, how does the trapped steam bubble behave?

ATTACHMENT #2 (Continued)

In addressing these issues, a review of the calculated plant response was performed in order to assess the controlling phenomena. As a result of that review, it was determined that the governing phenomena were:

1. Interruption in Natural Circulation

   - Spatial heat transfer in the steam generator
   - Distribution of steam flow from the core
   - Phase slip within the hot leg
   - Steam condensation in the steam generator

2. System Recovery Phase

   - Steam condensation on steam-liquid interface

Test data supporting the modeling of these phenomena has been evaluated and reported in the documents listed above. Further understanding of the plant response is provided in a qualitative assessment of plant behavior to various input and modeling assumptions contained in this report. It is clear that the concern on the interruption of natural circulation is a byproduct of the Appendix K assumption on HPI flow. Using the single failure assumption of Appendix K, it is shown in this report that phase slip modeling is important to the development of the plant response. Phase slip modeling is a part of the current SBLOCA Methods Program. The adequacy of current phase slip modeling was shown in the evaluation of test data discussed in the April 16 meeting with the Staff and summarized in this report.

ATTACHMENT #3

Responses to the Eisenhut to Mattimoe letter of March 25, 1982.

1. Interruption of Natural Circulation

   - Branch Flow

     The effect of preferential steam flow to the hot leg or the RV head has been addressed in the "Bubble Dynamic" report (see Attachment #2). Branch flow was discussed with the Staff in the April 16, 1982 meeting.

   - Hot Leg Flow Regime

     This was addressed in the Slip model presentation to the Staff on April 16, 1982 and is discussed in the report "Bubble Dynamics" (see Attachment #2).

2. Cold Leg Thermal Shock

   The concern over cold leg thermal shock was derived, as we understand, from TRAC computer calculations performed by Los Alamos National Laboratory (LASL) for the Staff, wherein significant cyclic temperature variations were shown in the vicinity of the cold leg emergency core cooling injection. We encourage the Staff to have an independent review performed on these calculations by an organization familiar with the hardware and components of the B&W designed system. If the cyclic behavior is confirmed, programs are already in place to address thermal shock and this item would be included in that effort.

3. Hydraulic Stability Following Accident Recovery

   This concern is addressed in the report "Bubble Dynamics" and was discussed with the Staff on April 16, 1982. In addition, the presentation given in that meeting, "Steam Condensation on Steam-Liquid Interface", also addresses the governing phenomenon in the recovery phase.

Other concerns in the March 25 letter were: break isolation, steam generator tube rupture, and cooldown and depressurization following a SBLOCA. These concerns are covered by the ATOG Guidelines and some are specific per plant type. Further discussion on these items is expected but not as a part of II.K.3.30.