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DENTON,H.R.	Office of Nuclear Reactor Regulation, Director

SUBJECT: Responds to NRC 810401 ltr advising util that proposed plans to upgrade & justify B&W small break evaluation model were acceptable in eight of nine areas.Current program adequate re technical needs & further test unnecessary.

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JF

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

July 24, 1981

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

Dear Sir:

By letter dated April 1, 1981, the Nuclear Regulatory Commission advised Duke Power Company that proposed plans to upgrade and justify the B&W small break evaluation model were acceptable in eight of nine areas. During a meeting on May 11, 1981, the Staff was appraised of the current schedule for completion of these items as well as informed of the scope of work undertaken in greater detail than was presented in my letter of January 30, 1981. The one item which has not been agreed upon is the need for overall model verification against integral system experimental data.

In your letter, you stated the NRC desire to require B&W licensees to verify the small break LOCA evaluation model with comparisons to integral system test data from a facility representing the B&W configuration. Since no such facility currently exists, you have suggested a cooperative effort to design and construct a facility which will provide the necessary information. This letter expresses our concern as to the appropriateness and fairness of this decision, our evaluation of the validity and capability of a test facility of the scale proposed to represent the B&W design--specifically the phenomena in question related to small break LOCA and certain transients, and foremost, the necessity of integral testing of these phenomena. An alternative to the NRC approach is suggested which will, in our estimation, provide for resolution of what constitutes a verified and conservative small break evaluation model.

Verification of the B&W small break evaluation model is an ongoing effort which has utilized integral system test data as applicable data becomes available. Recent comparisons have included LOFT tests L3-1 and L3-6 and Semiscale tests S-07-10B and D. Both of these facilities are representative of a PWR with U-tube steam generators. In comparison with these tests the B&W CRAFT 2 code has demonstrated its capabilities to predict the major phenomena associated with these types of transients. The code has also been tested by many predictive analyses of the B&W plant response to various small break LOCA and extended loss of feedwater transients. The results of these analyses have been very acceptable, and for those areas where the model could be improved, improvements have been undertaken. Several thermal hydraulic processes are predicted by the codes to occur during the transients of interest which are unique to the B&W design due to the OTSG. These processes have not been experimentally verified either in a plant or test facility. As an

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example, consider the effect of primary system repressurization/condensation following interruption of natural circulation. Although the integral effect has not been tested, the separate effects which when combined determine the integral effect, are well understood. The correct modeling of these separate effects along with an evaluation that the code predicted processes are physically realistic provide the necessary and sufficient verification of the capabiltiy of the evaluation model to handle the integral effects in question. This is a reasonable and attainable and justifiable method for resolution of the capability of the B&W code. This method has been utilized extensively in the past, and as the level of knowledge of transient processes has increased the codes have been reviewed and updated.

It is apparent that this method has been judged to be insufficient for review of the CRAFT 2 prediction of the response of the B&W plant to small break LOCA and related transients. As previously stated, the CRAFT 2 code has been verified by comparison with data from the LOFT and Semiscale configurations. This should support the validity of the equations and correlations in the code. The small break model improvement program that was presented to the Staff on May 11, 1981 contains several tasks where existing code models will be updated.

Historically, the LOFT and Semiscale facilities have been very beneficial to the nuclear industry for code verification. The tests have been performed in a very professional manner which reflects on the technical expertise and dedication of the staff and management at these facilities and their counterparts at NRC and DOE. The tests have provided excellent data in the areas of separate effects and have advanced the state-of-the-art in instrumentation. All test facilities are inherently limited in the ability to maintain scale with a commercial plant. This must be thoroughly considered in any extrapolation of LOFT or Semiscale data to commercial PWRs, and it has been in most cases. Unfortunately some scaling problems, particularly in Semiscale, cause integral effects which when understood are usually facility specific, and are not applicable to commercial plants. For severe LOCAs such as large break tests, the transient is rapid and scaling problems may be less evident and less important. Considering a small break LOCA with an extended period of two phase hydraulics and heat transfer and low flow conditions, the potential exists for problems of scale becoming major if not dominant forcing functions on the progression of the transient.

The objective of the proposed facility is to collect data on integral effects specific to a B&W configuration for small break LOCA and related transients. It is doubtful that any useful integral effect data is obtainable for small break LOCA code verification beyond that which can be obtained through the existing facilities. This is particularly true for the very transients which are of interest. The probability of that data being affected by scaling problems is considered to be rather high. What is likely is that data will be accumulated which will require a large effort to determine the facility specific responses recorded, and why the responses are not valid for a commercial plant due to scaling.

In summary, we feel that the program currently being undertaken in the area of small break evaluation model is adequate from the standpoint of technical needs and that further licensee effort in the form of integral tests is unnecessary.

Very truly yours,

William O. Parker, Jr.

RLG/php

cc: (w/copy of April 1, 1981 letter)

Chairman Palladino
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Director
Office of Management and Budget
Washington, D. C.

DUKE POWER CO.
PROJECT COORDINATION & LICENSING
APR 6 1981

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 01 1981

Mr. William O. Parker
Vice President, Steam Production
Duke Power Company
Power Building
422 S. Church Street
Charlotte, North Carolina 28202

Dear Mr. Parker:

In your letter to me on January 30, 1981, you advised NRC of your plans to upgrade certain parts of the B&W small break evaluation model and provide further justification for other parts of the model.

For the nine specific areas identified in your letter for upgrade or further justification, we believe that your proposed plans should satisfactorily address our concerns in eight of these areas. However, you indicate a completion date of March 1, 1982 for this effort, three months beyond the required submittal date of January 1, 1982 identified in NUREG-0737. Unless acceptable justification is presented for this delay, we will require you to complete this effort and submit the necessary information by January 1, 1982 as originally scheduled.

In Item 5 of your letter, you indicate nodalization studies will be performed. We have no objection to this effort, and in fact would encourage it. However, the staff concern expressed at the December 16 meeting (Task II.K.3.30 in NUREG-0737), to which this item responds, was not related to nodalization, but rather was with regard to the need for overall model verification against integral system experimental data. Specifically, the staff advised the representatives of the B&W Owners Group at the meeting that integral system two-phase natural circulation test data (i.e., representative of small break conditions) applicable to the B&W primary system design would be required for overall model verification. The staff further stated that because of the unique configuration of the primary system in B&W plants, we did not believe that two-phase natural circulation data from either LOFT or Semiscale was appropriate for overall verification of models to be used for evaluating B&W-designed plants. A more detailed description of the basis for the conclusion and the present staff concerns regarding small break behavior in B&W-design NSSS's are provided in the enclosure.

We consider this issue to be of considerable importance and believe that confirmatory experimental information is required. While we believe that the ultimate responsibility to provide this information lies with each applicant and licensee, we also recognize the significant costs that would be associated with such an endeavor. Therefore, we would like to discuss with the applicants and licensees with B&W-designed NSSS's the

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possibility of a cooperative program to construct a test facility that will provide the necessary information.

While formal Commission approval would be required for any cooperative government industry program of this magnitude, we believe that preliminary investigations between the NRC staff and industry can and should commence. NRC would be willing to share some of the costs of such a program through participation in the Semiscale Research Program at the Idaho National Engineering Laboratory. To initiate the effort, we suggest that an Owners Group be established, with B&W participation, to discuss with NRC the arrangements for such a cooperative program. NRC would be prepared to supply the building, support facilities, operating costs, and perhaps some of the other costs (e.g., analysis support) of the program. The design construction, and hardware costs for the test facility are estimated to be in the range of $15-20 million. We would expect that the details of the facility design and test program would be jointly worked out by the Owners Group and NRC.

Since our need for additional data, as well as our proposal for a cooperative program to obtain this data, is generic to all licensees and applicants with B&W-designed NSSS's, we are forwarding this letter to them as well.

You are requested to send a representative to NRC Headquarters for a meeting on April 23, 1981 to discuss this matter further. If you have any further questions regarding this letter or details of the proposed meeting, please contact Dr. Brian Sheron of my staff at (301) 492-9453.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Description, Conclusion and Staff
Concerns, Small Break Behavior,
B&W-Design NSSS's

BASIS FOR STAFF CONCERNS ON THERMAL-HYDRAULIC BEHAVIOR OF B&W PLANTS

As a result of the accident at TMI-2, the staff is reevaluating the capability of LWRs to accommodate small breaks in the primary system. In particular, we have focused on those small breaks which do not necessarily uncover the core, but produce two-phase conditions in the primary system and rely upon natural circulation to transport decay heat from the core to the steam generator.

Under these conditions, natural circulation flow is strongly influenced by energy transfer rates and flow path configuration, particularly piping elevations and location of condensing surfaces.

Because of this, predicted two-phase natural circulation behavior during small breaks in plants with NSSSs designed by B'W is uniquely different from the behavior predicted from plants with NSSSs designed by Westinghouse and Combustion Engineering.

Specifically, in plants with B&W NSSSs, natural circulation is predicted to be interrupted during the course of certain small break accidents due to steam accumulation at the top of the hot leg U-bends. In plants of the lowered-loop variety, this interruption is predicted to occur once, and then persist until the break flow causes the primary system to drain down sufficiently to expose a condensing surface in the steam generator (considered to be the elevation of the AFW spray sparger). During this interruption of natural circulation flow, decay heat removal is also interrupted, and a repressurization of the primary system is predicted to occur and persist until condensation (and subsequent energy removal) is restored. In plants of the raised loop variety (e.g., Davis Besse 1, Bellefonte), this interruption is predicted to occur three times, leading to three depressurization/repressurization cycles.

Another aspect of the B&W-designed NSSS is the relatively rapid response of these systems to secondary side upsets, particularly in the feedwater system. One aspect of ongoing efforts in this area is the assessment of methods which could alleviate, to some degree, this responsive behavior. A more complete and descriptive assessment of this concern is documented in Reference 1. Presently, any design changes and/or modifications proposed to reduce this sensitivity would be justified by analysis only, without the benefit of experimental verification (unless it could be safely demonstrated with plant testing). An experimental facility would be of significant benefit in this area of concern.

Based on the above concerns relating to the unique thermal-hydraulic behavior of B&W plants to both small break LOCAs and non-LOCA transients, and the uncertain ability of current analytical models to predict this behavior, we have concluded that integral system experimental data is necessary.

Ref. 1: "Transient Response of Babcock and Wilcox-Designed Reactors," dated May 1980.

Enclosure