MODEL DOCUMENTATION

For May 7, 1979, report, entitled "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant."

1. Introduction

Section 6 of the May 7, 1979, report[3] entitled "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," provides small break analyses performed to supplement the small break licensing submittals of references 1 and 2.

The analyses that were provided in reference 3 can be categorized into the following general topics:

a. The effect and/or need of auxiliary feedwater for small breaks.

b. Small breaks in the pressurizer.

c. Small breaks which are insufficient to remove all the decay heat via the break. These breaks address the concerns raised in the Michaelson report.[10]

The results of these analyses demonstrate that, with appropriate operator action, the ECCS in B&W's 177-FA NSS will control the consequences of small breaks within the requirements of 10 CFR 50.46 and Appendix K. This report documents and describes the small break models used in the analyses reported in reference 3.

2. Method of Analysis

The analysis method used for the evaluation of cases provided in reference 3 generally utilized the approved ECCS Evaluation Model described in Chapter 5 of reference 4, BAW-10104, Rev. 3, "B&W's ECCS Evaluation Model," along with the model modifications of reference 5. As dictated by reference 6, the Bernoulli correlation was used for subcooled flow rather than the modified Zaloudek correlation as proposed in reference 5. The following conditions and system responses were generally assumed in the evaluation:

a. The reactor is operating at 102% of a steady-state power level of 2772 MWt. Decay heat is based on 1.2 times the 1971 ANS 5.1 standard for infinite reactor operation.

b. The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis. The Bernoulli equation was used for the subcooled

portion of the transient, while Moody's correlation was used in the two-phase portion.

c. The leak is assumed to occur in the bottom of the RC pump discharge piping. This results in a loss of approximately 30% of the total injected HPI, once HPI is actuated.

d. No offsite power is available.

e. The reactor trips on low pressure at 1900 psia.

f. The safety rods begin entering the core after a 0.5 second delay from the time the reactor trip signal is obtained.

g. The RC pumps trip and begin to coast down coincident with the reactor trip.

h. Main feedwater pump trips and begin to coast down coincident with the reactor trip.

i. ESFAS signal error band is considered in the analysis to signal the actuation of the HPI system. This results in an ESFAS actuation pressure of 1365 psia.

j. One complete train of the emergency safeguards system fails to operate, leaving two CFT's and only one HPI and one LPI system available for pumped injection to mitigate the consequences of the accident.

k. The auxiliary feedwater (FW) system is assumed to be available during the transient. Its main function is to remove heat from the upper half of the steam generator during the initial stages of the transient. When the secondary side of the steam generator becomes a source of heat to the primary system, the assumption of auxiliary FW maximizes the energy that must be relieved.

l. The peak linear heat generation rate in the hot pin is the maximum allowed by the technical specifications.

m. Primary metal heating is included as per the requirements of Appendix K.

Differences to the general list of assumptions given above for specific and analysis cases are provided in reference 3 and are also summarized below.

CASE A - Small Breaks Without Auxiliary Feedwater for 20 Minutes
(Section 6.2.1 of Reference 3)

a. Two complete trains of the emergency safeguards system operate when the applicable setpoints are reached.

b. The auxiliary feedwater (AFW) system is assumed to be unavailable because of single failure considerations. At 20 minutes for the .01 ft^2 break, operator action is assumed to have taken to manually establish the emergency feedwater flow. Only one auxiliary feedwater system is utilized in the analysis.

CASE B - Small Breaks Withoug Auxiliary Feedwater
and 2 HPI's at 20 Minutes
(Section 6.2.2 of Reference 3)

a. The auxiliary feedwater is assumed to be unavailable.

b. Two trains of the HPI System are manually actuated at 20 minutes.

CASE C - Loss of Feedwater Resulting in a Stuck Open PORV
(Section 6.2.3.2.1 of Reference 3)

a. Loss of feedwater occurs at t = 0, and a coincident turbine trip is assumed.

b. The PORV opens at 6.0 seconds based on the TMI-2 data for this event. A leak area of 1.05 $in.^2$ (0.0073 ft^2), which is the PORV area, is used.

c. Offsite power remains available, which allows the RC pumps to operate throughout the transient.

d. The auxiliary feedwater becomes available 40 seconds after the loss of main feedwater. The auxiliary feedwater level is controlled to 30 inches on the SG secondary side.

e. The reactor is tripped at 12 seconds on high pressure. This time is based on data from TMI-2.

CASE D - Stuck Open PORV
(Section 6.2.3.2.2 of Reference 3)

a. Break is the PORV stuck open at t = 0.

CASE E - Small Break in the Pressurizer PORV With No Auxiliary Feedwater and Single Failure of the ECCS
(Reference 8)

a. The auxiliary feedwater (FW) system is assumed not to be available during the transient.

b. The ESFAS trip, including signal errors, occurs at a RC pressure of 1415 psia.

c. Operator action to initiate feedwater at 30 minutes or to establish a second HPI pump will prevent core uncovery.

CASE F - Small Break in the Pressurizer (PORV) With No Auxiliary Feedwater and Single Failure of the ECCS With Realistic Decay Heat
(Reference 9)

a. The reactor is operating at 102% of a steady-state power level of 2772 MWt. Decay heat is based on 1.0 times the 1971 5.1 standard for infinite reactor operation (realistic decay heat).

b. The auxiliary feedwater (FW) system is assumed not to be available during the transient.

c. The ESFAS trip, including signal errors, occurs at a RC pressure of 1415 psia.

3. Model Description

3.1 Category 1 and 2 Breaks

The CRAFT2 code was used to calculate the reactor coolant system hydrodynamics during the small break transient. The model used for the analysis of Category 1 and 2 breaks in the May 7th report utilizes the approved small break noding arrangement and is the same as that utilized in the generic 177-FA lowered-loop

plant small break analysis reported in the letter J. H. Taylor (B&W) to S. A. Varga (NRC), dated July 18, 1978.[2] The CRAFT model (see Figure 1) uses 20 nodes to simulate the reactor coolant system, two nodes for the secondary system, and one node for the reactor building. All breaks, other than breaks in the pressurizer, analyzed in this report are assumed to be located at the bottom of the cold leg piping between the reactor coolant pump discharge and the reactor vessel. Thus, after the HPI's are actuated, approximately 30% of the total injected HPI water is calculated to be lost via the break. The availability of auxiliary feedwater is case dependent as is discussed in the previous section.

3.2 Category 3 Breaks

The papter "Decay Heat Removal During a Very Small Break LOCA for a B&W 205 Fuel Assembly PWR" by C. Michaelson, dated January, 1978[10] states that, for very small breaks, the steam generator is necessary for heat removal. Should natural circulation be interrupted, primary system repressurization could occur and a potential exists for core uncovery for these very small breaks. The analysis provided in reference 3 demonstrates for the 177-FA lowered-loop plant, although system repressurization will occur for very small breaks, these breaks do not result in core uncovery.

The CRAFT2 model used in the analysis of the Category 3 breaks (Reference 8 and 9 and Section 6.2.4 of Reference 3) is shown in Figure 2. The model and input assumptions are basically identical to that used for the 177-FA lowered-loop plant small break analysis submitted in the letter report of July 18, 1978, from J. H. Taylor (B&W) to S. A. Varga (NRC).[2] However, additional nodes were added (nodes 24 and 25) to more accurately represent information of a steam bubble between the auxiliary feedwater injection point and the 180° bend in the top of the hot leg. The nodes contain 90° of the 180° bend in the hot leg, the hot leg volume between the 180° bend and the steam generator, the steam generator upper head, and the steam generator tube volume between the upper tubesheet and the elevation of the auxiliary feedwater injection nozzles. By isolating this

node from the steam generator nodes, a steam bubble can form within the upper portion of the RCS without being condensed by the steam generator. Also, the paths between the hot legs and the new nodes (broken loop paths 5 and 40 and unbroken loop paths 20 and 42) were added to allow for the occurrence of countercurrent flow in the 180^{o} bend.

The addition of two nodes within the CRAFT2 model were necessary in order to predict the RC system behavior for these smaller breaks. As demonstrated in section 6.2.1 of the reference 3, for breaks larger than 0.02 ft^2, heat removal via the steam generator is not necessary as the break plus condensation caused by the HPI provides sufficient heat sink. Thus, detailed modeling of the steam generator heat removal process is not necessary for these larger sized breaks. For the small breaks which exhibit repressurization and rely on SG heat removal, i.e. breaks less than 0.02 ft^2, a more detailed representation of the natural circulation phenomena is necessary to predict system response. Thus, the additional noding is required to provide a representation of system behavior during the transient.

1 BAW-10103A, Rev. 3, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," July 1977.

2 Letter from J. H. Taylor (B&W) to S. A. Varga (NRC) of July 18, 1978, concerning 177-FA plants small break analysis.

3 "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-Fuel Assembly Plant," May 7, 1979.

4 BAW-10104, Rev. 3, "B&W's ECCS Evaluation Model," August 1977.

5 Letter from J. H. Taylor (B&W) to S. A. Varga (NRC) of May 26, 1978, concerning proposed modifications to BAW-10104, "B&W's ECCS Evaluation Model."

6 Letter from S. A. Varga (NRC) to J. H. Taylor (B&W) of September 5, 1978, Subject: Evaluation of BAW-10104, Rev. 4.

7 BAW-10092, Rev. 2, "CRAFT2 - Fortran Program for Digital Simulation of a Multinode Reactor Plant During LOCA," April 1975.

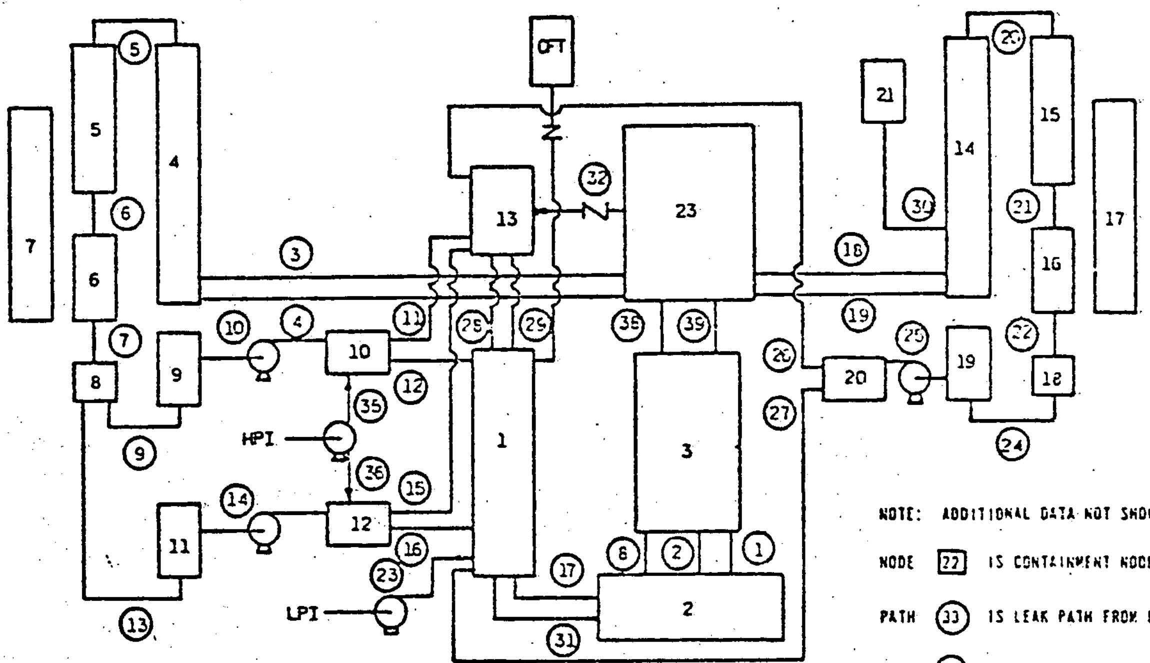
8 "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-Fuel Assembly Plant," Volume 1, Section 6.0 - Supplement 1, May 12, 1979.

9 "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-Fuel Assembly Plant," Volume 1, Section 6.0 - Supplement 2, May 12, 1979.

10 Paper, "Decay Heat Removal During a Small Break LOCA for a B&W 205 Fuel Assembly PWR," by C. Michaelson, dated January 1978.

FIGURE 1.

NODING DIAGRAM FOR SMALL BREAKS CRAFT2



NOTE: ADDITIONAL DATA NOT SHOWN ON DIAGRAM ARE:

NODE [22] IS CONTAINMENT NODE

PATH (33) IS LEAK PATH FROM BREAK TO CONTAINMENT

PATH (34) IS RETURN LEAK PATH FROM CONTAINMENT TO BREAK NODE

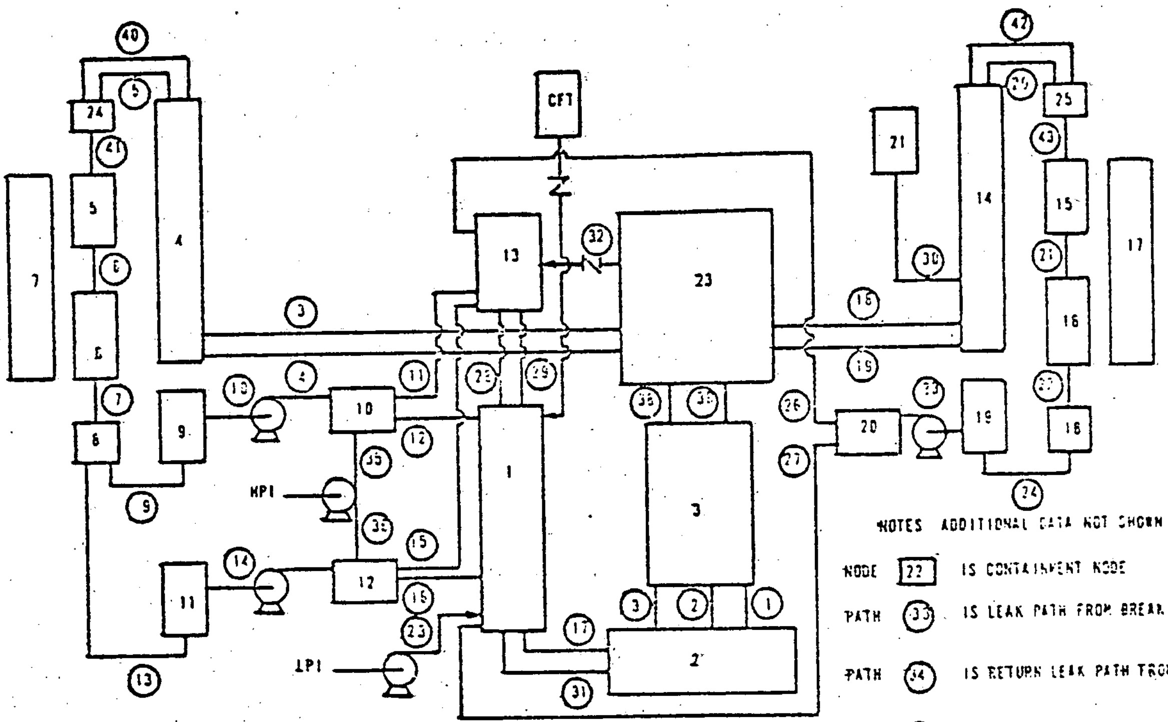
PATH (37) REPRESENTS CONTAINMENT SPRAY SYSTEM

NODE NO.	IDENTIFICATION
1	DOWNCOMER
2	LOWER PLENUM
3	CORE & UPPER PLENUM
4,14	HOT LEG PIPING
5,15	SG & UPPER HEAD
6,16	STEAM GENERATOR TUBES
7,17	SECONDARY, SG
8,18	SG LOWER HEAD
9,11,19	COLD LEG PIPING
10,12,20	COLD LEG PIPING
13	UPPER DOWNCOMER
21	PRESSURIZER
22	CONTAINMENT
23	UPPER PLENUM

PATH NO.	IDENTIFICATION
1,2	CORE
3,4,18,19	HOT LEG PIPING
5,20	HOT LEG, UPPER
6,21	SG TUBES
7,22	SG LOWER HEAD
8	CORE BYPASS
9,13,24	COLD LEG PIPING
10,14,25	PUMPS
11,12,15,16,26,27	COLD LEG PIPING
17,31	DOWNCOMER
23	LPI
28,29	UPPER DOWNCOMER
30	PRESSURIZER
32	VENT VALVE
33,34	LEAK & RETURN PATH
35,36	HPI
37	CONTAINMENT SPRAYS

FIGURE 2.

CRAFT2 NODING DIAGRAM FOR SMALL BREAKS



NOTES ADDITIONAL DATA NOT SHOWN ON DIAGRAM AREAS

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NODE NO.	IDENTIFICATION
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2	LOWER PLENUM
3	CORE
4,14	HOT LEG PIPING
5,15	SG & UPPER HEAD
6,16	STEAM GENERATOR TUBES
7,17	SECONDARY, SG
8,18	SG LOWER HEAD
9,11,19	COLD LEG PIPING
10,12,20	COLD LEG PIPING
13	UPPER DOWNCOMER
21	PRESSURIZER
22	CONTAINMENT
23	UPPER PLENUM
24,25	SG UPPER HEAD

PATH NO.	IDENTIFICATION
1,2	CORE
3,4,18,19	HOT LEG PIPING
5,20,40,42	HOT LEG, UPPER
6,21	SG TUBES
7,22	SG LOWER HEAD
8	CORE BYPASS
9,13,24	COLD LEG PIPING
10,14,25	PUMPS
11,12,15,16,26,27	COLD LEG PIPING
17,31	DOWNCOMER
23	LPI
28,29	UPPER DOWNCOMER
30	PRESSURIZER
32	VENT VALVE
33,34	LEAK & RETURN PATH
35,36	HPI
37	CONTAINMENT SPRAYS
41,43	SG UPPER HEAD