BW

B&W NUCLEAR TECHNOLOGIES

Rec'd 1/9/90

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JHT/89-256

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Ms. V. H. Wilson
Chief Administrative Section
PMSB
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: LOCA Evaluation Model Topical Report BAW-10168P

Reference: M. W. Hodges to J. H. Taylor, Request for Additional Information on BAW-10168P, RSG LOCA, December 1, 1988.

Dear Ms. Wilson:

Enclosed are revised responses to questions 8, 22, 24, 26, 27, 37, 40, 43, and 62 of the subject request for additional information. These responses are being revised as a result of discussions between the NRC, INEL, and B&W that have taken place over the last several months.

Very truly yours,

J. H. Taylor
Manager, Licensing Services

cc: w/o attach
R. B. Borsum
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w/attach
Gene Hsii, NRC

T010
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8. Question: Sections 4.3.2.9 and 4.3.2.10 of Volume I and Section 4.3.1.9 of Volume II referenced B&W report BAW-10091 to show that the effects of heat transfer from primary piping, vessels, and internals and secondary to primary heat transfer were minimal for LBLOCA and SBLOCA, respectively. Because this report was for B&W plants, provide additional information or analysis to justify the applicability of the conclusions of the report to the plants listed in Table 1-1, Volume I. Also, what are the criteria used to determine whether or not to model a metal mass?

Response: The LBLOCA and SBLOCA models contain primary metal slabs to properly simulate heat transfer from these metals. The study in BAW-10091 demonstrated that heat transfer from primary metals is not sensitive to conditions of surrounding fluids because the transfer rate quickly becomes conduction-limited. Furthermore, the amount of energy released from the primary metals and the steam generators for LBLOCA is small compared to core decay heat and flashing. Because the thicknesses of metal slabs in B&W, W and CE plants are comparable, the conclusions from the referenced study are applicable to the other designs. The primary metal model used in the evaluations contains all metal within or in contact with the reactor coolant system (RCS) water. Attached small piping, ECCS piping, instrument lines, and metal attached to the RCS metal are considered to have little impact on the LOCA results and are not included in the model.

The steam generators are important for both LBLOCA and SBLOCA as large reservoirs that act as either heat sources or as heat sinks. The energy of the pool of secondary coolant and the auxiliary feedwater has a profound effect on the transients. Within reasonable limits, however, the secondary metal and flow geometries do not substantially impact on the results of either the LBLOCA or the SBLOCA.

For the LBLOCA, the secondary system acts as a heat source after the first several seconds of the transient. Although unimportant during blowdown, the energy transport can easily vaporize and superheat the venting fluids during reflooding. That, in turn, worsens the steam venting and retards the plant flooding rates. During this period, the primary-to-secondary temperature differential is large and there is ample heat transfer area such that the details of the steam generator do not control the process. The only requirement is for the model to recognize and account for a large reserve of energy within the secondary system. The B&W evaluation model reflood simulation does this by setting the steam generator heat transfer coefficient conservatively high so that all incoming primary side fluid is vaporized and superheated to the secondary saturation temperature. Therefore, dependency on modeling detail and nodalization is removed from the simulation. Although of substantially lower importance, the amount of tube plugging modeled should be higher than that applicable to the plants to be covered by the analysis. At 10 to 20 percent plugging a difference of a few percent is not consequential, but differences of 5 percent or more will effect the results. The modelling within the B&W evaluation model for these parameters assures an appropriate and representative secondary inventory, sets a conservatively high reflooding secondary heat transfer coefficient, and employs a degree of tube plugging with selected margin.

For the smaller SBLOCAs, the steam generator acts as a heat sink. If the break flow cannot remove sufficient energy to keep the plant below the secondary pressure, the steam generator will absorb heat through steam condensation to maintain the primary system at a pressure near that of the secondary system. The available heat transfer area in the generator is large and the required heat transfer is small, so that the only substantial requirement on the modelling is to provide for the secondary side as a large reservoir of water. Of substantially lower importance is the action of the steam generators as heat sources for larger SBLOCAs and the flow restriction offered by tube plugging. The primary coolant flow rate through the steam generators is not high at any time that there is a potential for core uncovery, and only small frictional differential pressures result. The alteration of these pressure drops by a few percent because of the effect of tube plugging will not alter the static heads of liquid within the system. Similarly, because the flow through the steam generators is steam, at any time at which the core may be uncovered, the action of the steam generator as a heat source is to superheat the steam and not to vaporize entrained liquid. This effect should be modeled, but it does not have the importance of either the heat sink effect for SBLOCA or the heat source effect on reflooding for LBLOCA. The modelling within the B&W evaluation model for these parameters, is the same as that for the LBLOCA and assures appropriate and representative treatment.

For both large and small breaks, metal masses for the secondary side are included in the modelling. The steam generator tubes and the tube sheet are modelled with the primary system. Table 8-1 provides representative numbers for the initial, steady-state, energy content of the three major secondary metal structures (the shell, the downcomer shroud, and the steam separators) in contrast to the total energy of both the primary and secondary systems. The total energy of the secondary side metal is only 10 % of the total system energy. Of this 9 % is associated with the thick SG shell metal and only 1 % with thin secondary metals. The thin metal is further isolated from the primary system because it is almost all, 99 %, the steam separators and located in the steam dome of the generator. At this location it can act to superheat steam but cannot directly supply energy back to the primary system. Therefore, the modelling of the secondary side metals is not crucial to the prediction of the peak cladding temperatures for either large or small break LOCA. At a minimum, representative values for the shell metal, appropriate for the steam generator designs used by the plant being evaluated, will be included in the secondary side modelling for LOCA calculations.

The text within the evaluation model report that could be taken to mean that the secondary system is not important to the course of the LOCA transients will be modified to reflect the discussions in this response.

Table 8-1 Initial Energies of Secondary and Primary Systems

System	Initial Energy *	% of Total
Secondary Shell	59.0×10^6 Btu	9
Secondary Downcomer Shroud	0.1×10^6 Btu	0
Steam Separators	7.0×10^6 Btu	1
Secondary Fluid	141.3×10^6 Btu	21
Primary System **	467.6×10^6 Btu	69

* For the purposes of this table the reference conditions have been set at 228 F and 20 psia which is saturation temperature and pressure corresponding approximately to the containment conditions during reflood for a ice condensor containment.

** This value includes an approximation of the contribution of fission and decay heating up to the time of peak cladding temperature. The actual value for a LBLOCA could be lower by about 15×10^6 Btu and the value for a SBLOCA higher by about 35×10^6 Btu.

22. Question: In order to ensure that long term cooling can be established, Section 8 stated that the last calculation of the ECCS Evaluation Model is the demonstration that the timing of the operator action to assure core throughput flow is early enough to prevent boric acid crystallization within the core. Describe how the timing of this operator action is calculated.

Response: The timing of operator action to realign ECCS to supply water to the RCS hot legs is approximately 17 hrs. This action will provide a throughput flow and prevent boric acid crystallization for cold leg breaks. The timing is specified in plant technical specifications or in the operating procedures. Because the problem of particulate concentration is dependent on system design and reactor power and not dependent on fuel design it will generally be true that the previous basis for the technical specifications will remain valid for the BWFC fuel. In such cases a reference to the currently approved previous calculations will be provided. In the event that a customer wishes to alter a technical specification that might affect the timing of operator action, power level change or change in the ECCS systems, a calculation of the time dependent concentration of boric acid will be done to confirm or set the timing of operator action.

The following is an excerpt from the response to question 57 of this set which describes the concentration process.

> For plants with cold leg injection, the long term cooling mode following a cold leg break is one of extended core boiling. ECCS water is drawn from the reactor building sump, cooled, and injected into the plant cold legs from which it flows to the downcomer. At the downcomer, however, the hydrostatic balance between the core and the downcomer prevents most of the injection water from flowing to the core. Only that amount of water necessary to match the boiling flows to the core; the rest flows

into the broken cold leg, out of the break, and back to the sump. Water in the core is boiled, and steam is passed through the loops to the break and the containment where it is condensed by the building cooling system and returned to the sump as water. The procedure cools very well but can pose a problem. Boric acid does not vaporize at the same rate as steam, and most is left in the solution. (While most calculations assume all boric acid is held in solution, a good portion actually does vaporize.) This causes a gradual buildup in concentration of the boric acid within the core region. The process cannot start for the case of hot side breaks as the ECCS always flows through the core for those events and the distillation process never starts.

Although the solubility saturation limit for boric acid is considerably above the solute concentration of the injection water, the process, unchecked, could cause high enough concentrations to lead to precipitation of borate crystals. The injection solute is about 1.22 weight percent boric acid (this number will vary by plant) and the solubility limit at $212^{o}F$ is slightly over 35 percent. Figure 57-1 shows the saturation solubility of boric acid as a function of temperature. The figure is derived from reference 57.1 and was presented as part of the B&W evaluation model for B&W designed NSS in reference 57.2, BAW-10091 Supplement 1 page 3-90. The most common procedure is to establish a small amount of hot leg injection which, once larger than the core boiling rate, will cause the flow in the vessel to reverse and proceed from the core through the downcomer and out the break. The flow from the core to the downcomer will be at the core concentration and act to remove boric acid from the system. This leads to the eventual control of the concentration process and gradual decrease in the core concentration of boric acid.

To calculate the effect of this process a simple two-step concentration calculation is done. The first step initiates at the time of the accident. The mixing region is taken as the core and the upper plenum up to an elevation equal to the bottom of the cold leg pipe. The concentration of boric acid is considered uniform within the mixing region with the initial concentration being equal to that of the initial ECCS injection concentration. Water is removed and added to the system at the rate of core boiling. Boric acid is added to the mixing region with the replacement water at the concentration in the downcomer. The downcomer concentration is initially that of the initial ECCS but after the start of sump recirculation it will deplete as the core concentration builds. No boric acid carry over with the vaporized water is credited (the expected carry over concentration is about 300 ppm). This step ends with the operator action to start core flow through.

Depending on the plant the second step adds the modeling of some degree of core throughput flow to the process. Usually this will be a small amount of hot leg injection. To the degree that core boiling remains, water is still removed and boric acid left behind. However, to the degree that the new injection is not fully boiled, water is removed from the mixing region along with boric acid at the concentration of the mixing region. As boiling decreases with decay heating the concentration eventually peaks and starts to fall back to the average concentration of the initial ECCS injection.

So long as the peak concentration remains below the saturation limit (see Figure 57.1) the timing of operator action is adequate. Should a revision to the operator action be necessary, the calculations are repeated assuming various timings until an acceptable action time is identified. The amount of assured core throughput flow can also be varied to allow differing operator action times.

24. Question: Figure 4-1 shows the LBLOCA code interfaces in the B&W EM methodology.

a. The figure shows that RELAP5/MOD2-B&W is used to calculate the system response from time zero to the end of blowdown (EOB). The resulting core enthalpy, pressure, and mass flux from RELAP5/MOD2-B&W are then passed to FRAP-T6-B&W which calculates the hot rod response from time zero to the end of the adiabatic heatup (EOAH). Because the time of EOAH is greater than the time of EOB, there is a period where the required boundary conditions for FRAP-T6-B&W are not provided. Clarify the code boundary condition requirements for FRAP-T6-B&W during the period in question and discuss how they are provided in the EM methodology.

Response: Two key parameters affect fuel pin heatup during the adiabatic heatup period (from the EOB to the beginning of reflood): (1) core decay heat and (2) clad outside surface heat transfer coefficient. The core decay heat used is discussed in BAW-10168 Volume I Section 4.3.2.1 [24.1]. It consists of 120% of the 1971 ANS proposed standard plus allowance for actinide decay and continued fission power. The surface heat transfer coefficient is set to zero during this period.

b. The figure also shows that core parameters at the end of adiabatic heatup are passed from FRAP-T6-B&W to BEACH. Clarify which core parameters are passed.

Response: The core parameters (at EOAH) passed from FRAP-T6 to BEACH are fuel rod temperatures and fuel pellet-to-clad gap conductance for each axial node.

c. This figure and Figure 4.1 of Volume II regarding SBLOCA code interface do not show any mechanism for possible feedback between the codes that calculate the system response to the same portion of the accident. How are differences in the code results handled? For example FRAP-T6-B&W calculates the hot rod response from the beginning of the accident to the end of the event. RELAP5/MOD2-B&W and REFLOD3B also calculate portions of the accident calculated by FRAPT6-B&W. If the FRAP-T6-B&W analysis shows more flow blockage due to rod swell and rupture than the REFLOD3B or RELAP5 results, how is this type of difference handled? For SBLOCA, specifically clarify how the differences in the mixture levels calculated by RELAP5/MOD2-B&W and FOAM2 are handled.

Response: There are no duplications of roles in the B&W EM code package shown in Figure 4-1 of Volume I or Volume II, and the limited exchange of data between codes is better characterized as time-dependent boundary conditions than as feedback. The interactions are arranged in series through the use of a hierarchy for the computation of certain events or phenomena. The application of each code is designed such that the results of the combined applications are more conservative than those that could be achieved by a fully integrated application.

Although not an expected event for recirculating steam generator plants, an example of the hierarchy occurs in the handling of a blowdown rupture. In such a circumstance both RELAP5/MOD2-B&W (RELAP) and FRAP-T6-B&W (FRAP) would calculate the occurrence of rupture within the hot assembly. Although the results of these two codes tend to agree reasonably well with each other, they can not be expected to coincide exactly, and the following procedure applies: The timing of rupture and the cladding strain effects, on heat transfer and oxidation rate for the peak cladding temperature calculation, are taken from the FRAP-T6 prediction. RELAP5 accounts for the effects of rupture --flow blockage, cladding strain (which increases metal water reaction), and the addition of interior oxidation --on the fluid state. This is appropriate because the RELAP5 role is to provide fluid boundary conditions to FRAP-T6 and not to calculate the final cladding temperature. Because of the heat transfer simulation, RELAP5 will generally calculate a rupture earlier than would FRAP-T6. In this case the timing of the rupture in RELAP5 will not be adjusted to agree with FRAP-T6 and the resultant post rupture flow decrease is treated as a conservatism. Should the reverse occur and FRAP-T6 predict a rupture prior to RELAP5 then the time of rupture in RELAP5 will be forced to coincide with FRAP-T6 and RELAP5 restarted for the remainder of blowdown. This exchange of data occurs once in a run and is considered as a time-dependent boundary condition.

The rupture induced strain for the purpose of determining flow blockage is recognized as different from the strain used to compute cladding heat transfer areas and oxidation surfaces. NUREG-0630[24.2] provides separate curves for these effects. The only code in the B&W evaluation model to calculate a flow blockage effect is RELAP5/MOD2-B&W. REFLOD3B evaluates only the average core and no provision is made for rupture within the code. Flow diversion away from the rupture location is treated (bounded) by applying the average channel flooding rate to the hot channel temperature calculation. Radial effects during reflooding, primarily the greater conversion of water to steam in the higher powered assemblies, promote higher effective flooding rates in the hot assemblies than in the colder assemblies. This has been observed the CCTF and SCTF facilities, which are large enough to incorporate radial power shaping. Similar trends are observed in these facilities for asymmetric power and initial temperature tests. Additionally, as provided in the response to question 53 of this set, the FLECHT-SEASET tests with simulated ruptures support a conclusion that if cross bundle flow diversion does exist it is not of consequence. Therefore, the use of the average channel flooding rate for the hot assembly bounds the effect of rupture induced crossflow. The treatment of rupture effects during the reflood period for the hot channel cladding temperature differs between the original and revision 1 of the evaluation model.

For the original evaluation model, FRAP-T6-B&W, as discussed, uses boundary conditions, which include the effects of flow diversion to compute the hot spot temperatures. In doing so FRAP-T6 must adjust the local cladding-to-fuel gap and cladding surface area, upon rupture, according to the NUREG-0630 recommendation for strain to be used to compute cladding heat transfer areas and oxidation surfaces. BEACH is restarted at the time of rupture with its gap coefficients and cladding surface areas set equal to those predicted by FRAP-T6 just following rupture. As with the possible forcing of rupture in RELAP5, this exchange of information is considered a time-dependent boundary condition.

For revision 1 of the evaluation model the FRAP-T6 code has been replaced by an upgraded BEACH version during the reflooding calculation. This required the addition of full gap and rupture simulations to the BEACH coding. Therefore, because the calculation for cladding temperature is performed within a single code there is no longer any occurrence of a time-dependent boundary condition during the reflood portion of the transient evaluation.

The hierarchy for mixture height and liquid inventory within the reactor vessel during small break LOCA is as follows: The core liquid inventory used in the FOAM2 code is taken from RELAP5/MOD2-B&W. The mixture height and steam flows, as calculated by FOAM2, are used in FRAP-T6-B&W to compute hot spot cladding temperatures. Because the RELAP5 mixture level prediction may not agree with the FOAM2 prediction, experience and test FOAM2 runs are used to determine when the core liquid inventory has decreased to an amount at which core uncovery can occur.

26. Question: Each of the codes used in the LBLOCA and SBLOCA EM methodologies have a number of user inputs and option selections. To ensure that each code is used within its capabilities and with proper input, information justifying all selected options and input data, including defaults, should be presented. List all options, with bases for the chosen options, to be used for LBLOCA and SBLOCA analyses.

Response: The documentation of the code options in a list is not the way B&W has selected to control the evaluation model. As presented in 10 CFR 50.46 "An evaluation model is the calculational framework for evaluating the behavior ... It includes ... all information necessary for application of the calculational framework ... such as mathematical models used, assumptions ... , procedure for treating input." Thus, the evaluation model is a guide to the calculation sufficient in detail to assure that application of the guide accomplishes the intended calculational procedure.

In order to simplify understanding and maintenance of the evaluation model, the model reporting or documentation has been tightly controlled to eliminate to the extent possible duplication of information. Methods controlled by the evaluation model are presented in context and on as generic a basis as possible. Thus, the description of the flattening factor for the power at the hot spot is only provided once while it is incorporated in three codes; RELAP/MOD2-B&W, FRAP-T6-B&W, and BEACH.

B&W will provide specification of the options to be used in the calculations within context in the evaluation model report and to the extent necessary to assure that the proper and intended options are used. B&W will not provide documentation for options or approaches not to be used. It is, however, apparent that in the original release of BAW-10168, RSG LOCA, the B&W evaluation model report for recirculating steam generator designs, some of the information required was not included. To speed review a list of that information is provided as Table 26-1. The list is organized according to the documentation and will be used to update the evaluation model report in the future.

One specific type of option deserves unique treatment. At several locations throughout all of the computer codes, correlations have been programed to include user input multipliers. This was done to make provision for sensitivity studies and in some cases to allow for code applications to problems other than those of 10CFR50.46. It is not the intent of the evaluation model to allow variation in these constants. Therefore, unless specifically stated to the contrary in the evaluation model report these constants shall all have a value of one. Programed constants that are part of the correlations or part of B&W's implementation of the correlations will have the value as published in the individual code topical reports. The evaluation model will be revised to include these statements.

Inputs used during an evaluation can be categorized as follows:

Generic: User supplied values or constants whose values are controlled by the evaluation model. The materials properties are a good example of this type of input.

Prescribed: Input for which a determining procedure is specified in the evaluation model without the specification of a value. The use of hot fluid volumes within the RELAP5/MOD2 model is a good example of this type of input.

Plant:	Input which is taken from documentation for the individual plant or plants to be covered by the evaluation. Plant geometry inputs are examples of this type of input.
Case:	Input which will vary depending on the accident being evaluated. Break area is a good example of this type of input.

The B&W evaluation model controls these different types of inputs in differing ways. Generic and Prescribed inputs are controlled in the same fashion as code options, and are documented in context within the evaluation model report. As with the code options, it is apparent that the original release of BAW-10168, RSG LOCA, the B&W evaluation model report for recirculating steam generator designs, does not contain all of the information required on generic inputs. To speed review a list of that information is provided as Table 26-2. The list is organized according to the documentation and will be used to update the evaluation model report in the future.

Plant input is controlled by B&W internal calculational procedures and not by the evaluation model. These procedures are written to adhere to ANSI quality assurance standards. For the most part these procedures require that the input come from controlled design documentation, that it be referenceable, that its use be documented, and that an independent review be conducted to assure that this has been done. The documents attesting to this for any given evaluation are controlled documents stored at B&W and available for audit at any time.

Case input is much akin to assumptions. It is not controlled by either the evaluation model or B&W procedures other than that it must be documented along with the plant input in controlled stored records of the calculation. Case input is also available for audit at any time.

Table 26-1 Additional Evaluation Model Guidelines

Code Options Used in Evaluation Model

OPTION	SELECTION
RELAP5/MOD2-B&W	
Flow Film Boiling Lock-in	No lock-in until $T_w - T_s \geq 300$ F
Fine Mesh Rupture Option	Not used
Concentric or Non-concentric Fuel Pellet Simulation	Use Concentric with TACO2 Use Non-concentric with TACO3
Critical Flow Model	Subcooled - Ext. Henry-Fauske Two-phase - Moody Superheat - Murdock-Bauman
Leak Flow Slip Model	LBLOCA - Moody Slip SBLOCA - No Slip
Criteria for Break Discharge Coefficient	LBLOCA - No change in coefficient during run SBLOCA - coefficient changes with leak void fraction
Internal Critical Flow	Flow not limited except for special paths: downcomer to reactor vessel upper head bypass path, pressurizer surgeline, & accumulator lines
Friction	Calculated by RELAP5
Homogeneous-Equilibrium	Only in the core region
Heat Transfer Model	Core model is used for the active core heat structures the System model is used elsewhere
CHF & Film Boiling Correlation	The correlation combination with BWCMV for high pressure and high flow is used. Condie-Bengston IV is selected for flow film boiling

Table 26-1 Additional Evaluation Model Guidelines

Code Options Used in Evaluation Model - Continued

OPTION	SELECTION
RELAP5/MOD2-B&W	
Metal Water Reaction Model	Baker-Just
Rupture Temperature	Instantaneous ramp rate during pre-plastic region; Plastic weighted, time averaged ramp rate thereafter
Accumulator Wall Heat Transfer	Yes

Table 26-1 Additional Evaluation Model Guidelines

Code Options Used in Evaluation Model - Continued

OPTION	SELECTION
FRAP-T6-B&W	
Flow Film Boiling Lock-in	No lock-in until $T_W - T_S \geq 300$ F
CHF & Film Boiling Correlation	The correlation combination with BWCMV for high pressure and high flow is used. Condie-Bengston IV is selected for flow film boiling
Metal Water Reaction Model	Baker-Just
Rupture Temperature	A plastic weighted, time averaged ramp rate is used
Licensing Audit Code Models	All of the Licensing Audit Code Models are used except for: Cladding Specific heat, Elastic Modulus, Fuel Deformation, Operating Power x 1.02, and ANS Decay Power x 1.20
Clad Deformation	No Balloon Option, clad failure occurs per rupture curves

Table 26-1 Additional Evaluation Model Guidelines

Code Options Used in Evaluation Model - Continued

OPTION	SELECTION
REFLOD3B	
CRF / Core Heat Transfer	CRFCKN option: CRFCKN correlation for carry out coupled with FLECHT/ANC correlation for core heat transfer
Core Bypass	The core bypass model is selected
Vent Valve Model	May be selected by user to simulate downcomer to upper head bypass flow. Vent valve steam condensation efficiency in Equation 2-68.4 of the REFLOD3B topical is set to 1.0

Table 26-1 Additional Evaluation Model Guidelines

Code Options Used in Evaluation Model - Continued

OPTION	SELECTION
BEACH	
Friction	Smooth pipe option
Reflood	Reflood option selected

Table 26-1 Additional Evaluation Model Guidelines

Code Options Used in Evaluation Model - Continued

OPTION	SELECTION
FOAM2	
Two Phase Slip Corr.	Wilson

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

INPUT	SELECTION
RELAP5/MOD2-B&W	
Fluid Volumes	Hot - from design drawings
Attached Piping Volumes	Only the accumulator line volum[illegible] are included. The tota[illegible] other attached piping volu[illegible] lies within the accuracy of the system volume calculation and are not included
Initial Reactor Coolant System Flows	The system flows are those used in the at power minimum DNB analyses. The hot and cold leg temperatures are set by nominal control system response to that RCS flow
Two Phase Pump Degradation	Use Table 2.1.5-2 BAW-10164
Transition Quality for Single to Two-Phase Flow Correlation	0.0
Moody Slip Parameters	$0.5 \leq$ slip ratio ≤ 2.0
Leak Flow Smoothing	LBLOCA - $-0.001 \leq X \leq 0.001$ SBLOCA - $0.01 \leq \alpha \leq 0.70$
SBLOCA Discharge Coefficient Switch	Cd = 1.0 $\alpha \leq 0.70$ Cd = 0.7 $\alpha > 0.70$
Initial Inventories for Reactor Coolant System, Secondary System, and ECCS Systems Pressure	Set by nominal operation design levels. The volumes for attached piping except for the pressurizer surgeline are not included in the LOCA model
Primary Metal	Structures are lumped together according to material properties, thicknesses, and location. Grouping is user controlled

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

- Continued -

INPUT	SELECTION
RELAP5/MOD2-B&W	
Over Power Factor	2% applied to both average and hot assembly powers
Power Flattening	5% of the heat generated in the hot assembly is deposited directly in coolant. This factor is not applied to the average core
Decay Heat	120% ANS 1971 based on 102% core power
Actinide Heating	Fit to envelope of 1979 ANS 5.1 Standard
Initial Fuel Temperatures	Adjusted to agree with an NRC approved steady-state fuel performance code (TACO2/TACO3)
Rupture Data	NUREG-0630 ramp rate dependent data
Time Step Control Option	Option 3, mass error checking, consistent hydrodynamic and heat structure solution time advancement
ECCS Fluid Temperatures	Set at nominal year average temperatures per system
ECCS Time Delays	Includes provision for signal, diesel start-up, pump start-up, and line filling for all evaluations
Containment Pressure	Set from FSAR

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

- Continued -

INPUT	SELECTION
FRAP-T6-B&W	
Power Flattening	5% of generated heat deposited directly in coolant
Over Power Factor	2% applied
Power Distribution	Appropriate for the average fuel pin in the highest powered fuel assembly in the core. Axial peaking is the same as for RELAP5/MOD2-B&W evaluation and the radial peak is the relative hot assembly power
Decay Heat	120% ANS 1971 based on 102% core power
Rupture Data	NUREG-0630 ramp rate dependent data
Initial Fuel Temperatures	Adjusted to agree with an NRC approved steady-state fuel performance code (TACO2/TACO3)

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

- Continued -

INPUT	SELECTION
REFLOD3B	
Unrecoverable Loss Factors	Input separately from friction loss factors
Loss Factors Source	RELAP5/MOD2-B&W steady-state prediction
Fluid Volumes	Hot - from RELAP5/MOD2 model
ECCS Time Delays	Includes provision for signal, diesel start-up, pump start-up, and line filling for all evaluations
Containment Pressure	Set from FSAR
Decay Heat	120% ANS 1971 based on 102% core power
RC Pump Resistance	Appropriate for locked rotor condition
ECCS Fluid Temperatures	Set at nominal year average temperatures per system
Primary Metal	Primary metals are included with structures lumped together according to material properties, thicknesses, and location. Upper head metal is modelled in the RCS hot legs. Grouping selections are user controlled
Film Coefficients for Non-Core Heat Slabs	Covered portion of Reactor Vessel 200 Btu/Hr-ft^2-F Uncovered portion of Reactor Vessel 20 Btu/Hr-ft^2-F Remainder of the Primary System 1000 Btu/Hr-ft^2-F

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

- Continued -

INPUT	SELECTION
REFLOD3B	
Cold Leg Steam Water Interaction Pressure Drop	0.85 psia during accumulator injection and 0.5 during pumped injection
System Initialization	Core and upper head regions are saturated steam, hot legs and steam generators are superheated steam
Constants for CRFCKN	$C_{SUP} \geq 1.025$ ( ) , $C_T = 1.0$ (1/s) , $C_H = 1.0988$ (1/ft)
Condensation efficiency for Steam Vented Through the Upper Head Spray Nozzles	1.0

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

- Continued -

INPUT	SELECTION
BEACH	
Form Loss Factors	From assembly tests
Power Flattening	5% of generated heat outside of fuel pin
C_{maxDB}	Use value from Equ. 2.1.3-105.1 of BAW-10168 Rev. 2
Correlation for F_{gq}	Use the Yao, Hochreiter, and Leech correlation as described in Section 2.1.3.7.1.of BAW-10168
Grid Atomization Factor	Use Equation 2.1.3-99 of BAW-10168 revision 2 with n equal to 2.7
Fuel Temperatures	Initialized to FRAP-T6 results at the end of adiabatic heat up
Gap Multipliers	Initialized to those used by RELAP5 to obtain the initial (time 0) hot channel fuel temperatures

Table 26-2 Additional Evaluation Model Guidelines

Generic and Prescribed Inputs for the Evaluation Model

- Continued -

INPUT	SELECTION
FOAM2	
Steam Generation from Reactor Vessel Metals	Yes
Steam Generation from Flashing	Yes

27. Question:

a. For the crossflow resistance sensitivity study discussed in Section A.3.4, the base crossflow resistance was multiplied and divided by 100 to determine the sensitivity of the calculated results. The study showed essentially identical results for the three cases. Do you intend to use the base case crossflow resistance in all future EM calculations? If so, identify the resistance factors used in the base case study. Justify the applicability of these resistance factors to all fuel bundle types to be analyzed.

Response: The crossflow resistance factor used in the evaluation model studies expressed in unitless form is 50.0 based on a flow area of 1.3 square feet. This resistance factor was calculated from a correlation developed by Babcock & Wilcox and documented in Appendix H of the CRAFT2 code topical report[27.1]. The crossflow resistance is given as a correlation of data measured at the B&W Alliance research facility for flow between adjacent subchannels and is based on the velocity upset or mismatch between the assemblies. The resistance correlation is given as a function of fuel assembly pitch-to-fuel pin diameter, the number of fuel assemblies in the regions for which the crossflow is modeled, and the ratio of the Reynolds numbers in the regions.

The pitch-to-diameter ratio will vary only slightly with fuel type. That is, for all 17-by-17 fuel assembly designs, even the OFA (Optimized Fuel Assembly), this ratio is nearly the same. For differing fuel assembly types the ratio does not change substantially but it should be updated. B&W will use the base cross-flow resistances for all 17-by-17 applications and update the resistance for applications to alternate fuel types such as 14-by-14, 15-by-15, or 16-by-16.

The number of fuel assemblies in the adjacent regions is a function of the noding detail selected for the model. As this is fixed by the evaluation model, this determinant for the resistance will not be altered from plant to plant.

The ratio of Reynolds Numbers is determined by the initial flow relationship between the cross-flowing regions. As the hot channel initial flow differs from the average channel flow only by the amount of the chimney effect, the difference in Reynolds Number ratio between plants will be inconsequential.

Combining these considerations the deviation in cross-flow resistance from one plant and fuel design to another will be small. In particular, these differences will be much less than those imposed by the crossflow sensitivity study. Therefore, B&W will employ the base crossflow resistance for all applications, updated only for a major change in fuel type such as the difference between 15-by-15 and 17-by-17 fuel.

b. Appendix K, Item I.C.7.a, requires the effects of crossflow between hot and average channels be considered. Although a crossflow resistance study was performed, it only considered the initial crossflow resistance. The effects of rod swell and rupture on the crossflow resistance were not considered in this study, nor were the effects discussed in the sections on the rod swell and rupture model. Clarify how the effects of rod swell and rupture are included in the crossflow resistance.

Response: The effects of rod swelling and rupture within the core region are addressed by the core model if and when rupture occurs. The core flow model is a network of axial and radial flow paths connected at the core nodes. Upon the occurrence of rupture, the flow resistance for the axial flow paths connected to the node within which the rupture has been calculated are increased. Although the modelling is general the rupture will most likely occur in the hot channel. This results in a revised resistance network that can interact appropriately with the fluid properties as they exist. Depending on the core flow conditions at the time of rupture, the additional resistance to flow through the ruptured area could cause significant flow diversion away from the hot channel at the elevation of the node upstream of the rupture and back into the hot channel at the elevation of the node downstream of the hot channel. Experience shows, however, that the core conditions are typically such that the diversion of flow out of the hot channel is limited.

The modelling neglects the effect of adding resistance to the crossflow paths. As with flow resistance for axial flow, the resistance to flow radially will be increased as ruptures appear in the hot assembly. This increase in flow resistance to radial flow would tend to retain some coolant within the hot assembly near the locations of the ruptures which would otherwise be passed radially out of the hot assembly. The BWFC evaluation model makes the conservative simplifying assumption that the resistance to flow in the radial dimension is not affected by the occurrence of rupture. Therefore, the model tends to promote flow away from and out of the hot assembly.

No provision is made to increase the flow resistance in an axial segment because of the effects of prerupture strain. These strains result in slight bulges in the cladding surface that intrude into the flow channel. The geometry is prerupture and is therefore similar to a nozzle with a well rounded inlet and a well rounded exit. Such a configuration would not add substantially to the flow resistance for the degree of strain that occurs prior to rupture.

The study for crossflow resistance sensitivity included in the evaluation model report did provide for the effects of swelling and rupture. Those events, however, were not calculated to occur during blowdown. Rupture will occur for these runs during reflood, and the resultant flow diversions, if any, are allowed for by conservatisms in the BEACH code. Because core cooling during reflood is limited for recirculating steam generator plants, the probability that a calculation during which a blowdown rupture occurs can meet the criteria of 10 CFR 50.46 is small. Thus the sensitivity study evaluated the most probable circumstances for application of the evaluation model.

References:

27.1 CRAFT2 - Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant, BAW-10092-A Rev. 3, July, 1985.

37. Question: Appendix A discussed the results of nodalization studies for RELAP5/MOD2-B&W and REFLOD3B but did not provide the results of similar studies for BEACH and FRAP-T6-B&W. Provide the results of nodalization studies for these codes to demonstrate solution convergence with the nodalizations B&W intends to use in EM analyses.

Response: The BEACH and FRAP-T6-B&W codes are not subject to convergence problems caused by insufficient noding in the manner that RELAP5 and REFLOD3B are. For BEACH and FRAP-T6-B&W the noding arrangements used are governed by the physics modelled in the codes and the degree of detail desired in the answer. For both codes, increases in noding merely refines the average answers received from noding arrangements of lesser detail. Both BEACH and FRAP-T6-B&W employ approximately 21 axial nodes in the evaluation model. This gives an average node height of about 0.6 feet. Resolution of the temperature to a smaller region of the fuel would not alter the results and would only incur unnecessary expense. The following paragraphs present the rationale for the noding detail used in the EM applications of the codes.

The noding arrangement to be used in the BEACH code is not variable in the sense that RELAP5 is. During development of the BEACH code a given arrangement was selected and compared to data for justification. The noding used in any application must comply with that arrangement for the results to be considered as valid results of the BEACH code. The arrangement and approach, which may not dictate the absolute number of nodes, is presented and justified in Appendix C of the BEACH topical report. Question 18 of the first round of questions on the BEACH topical report, BAW-10166, requested further information on noding arrangements as does this question and the response is repeated in part here.

BEACH axial and radial noding is based directly on the basic modeling physics selected for the evaluation of the reflooding process. Although some experiments, such as the CCTF tests, have shown that radial effects within the core during reflood promote extra cooling in the hotter regions, the B&W Evaluation Model simulation of this process is one-dimensional. The BEACH model is a single stack of fluid volumes oriented vertically and representing the average hydrodynamic conditions within the hot fuel assembly. Under this assumption, the boundary conditions for BEACH are taken from the REFLOD3B simulation of the average core reflooding. The addition of a second radial channel to BEACH would only replicate the present solution unless the driving boundary conditions were to include radial variability. Thus, until a more sophisticated approach is adopted there is no need to conduct radial noding sensitivity studies for the BEACH code.

Axially, the noding is again selected in accordance with the physical modeling. In this case, the noding is tied to the grid models and the distribution of grids within the fuel assembly being studied. All present B&W-designed fuel assemblies would call for the same axial noding as was used in the FLECHT benchmarks. The axial noding, detailed in Appendix C of the BEACH topical report BAW-10166, is a series of seven, three-node groups. As each of the three nodes within an axial group experiences a different cooling pattern, it is necessary to use three nodes. Additional nodes, however, do not substantially alter the results.

The response to BEACH topical question 18 continues to describe some of the alternate noding schemes that were used at one time or another during BEACH development, what was observed from each, and how they lead to the current 21 node approach. The response finishes with:

It is concluded from this series of models that:

1. The three regions of grid effects on heat transfer must be modelled by individual nodes.

2. Once each region is modelled, there is little change to the results with additional nodes.

3. The 21-node grid model is adequate for LOCA evaluation model predictions.

Although 21 nodes were used in the benchmark evaluations, the requirement is to use at least a three-node pattern between grids for grid spans near the location of peak cladding temperature. The 21-node model is just the result of applying such a pattern uniformly over an assembly with seven grid spans. The modelling of the assembly below the first interior grid or above the last interior grid need not follow the three-node pattern. These regions of the assembly, being on the very ends are well removed from the locations of peak cladding temperature, may be modelled with fewer or more than three nodes. The selection is made by the user of the evaluation model and is usually based on a desire to preserve an approximately consistent node height along the fuel assembly. Between the lowest and highest interior grids the user must employ a three-node pattern per grid span. Reference to Appendix C of the BEACH topical and to the response to question 18 of the first round of questions on that topical can be made for further detail.

Because of the grid induced heat transfer pattern during reflood, the FRAP-T6-B&W noding for large break evaluation model applications has been selected to correspond to the BEACH noding. This scheme is also acceptable for the blowdown and adiabatic heatup periods because axial variations in boundary conditions within the resultant, 0.6 foot, node height are not strong and axial heat conduction insignificant. Therefore, no noding studies are indicated for the FRAP-T6-B&W code.

For small break evaluations the prediction of cladding temperatures at 0.6 foot increments is sufficient for the purposes of compliance with 10 CFR 50.46. The motivation here is more one of model assurance than of heat transfer performance. Grid effects are not considered to be dominant in the steam only flow that occurs during core uncovery for small breaks and the three node per grid span rule is not required. Using the same model as for large breaks, however, may result in some economies of model preparation and limit the opportunities for erroneous inputs. Therefore, the same noding as for large breaks will be used when the core uncovery transient lasts for under 500 seconds. For small break LOCA evaluations with core uncovery times in excess of 500 seconds nodes which continually lie below the mixture height may be combined to provide faster running times.

40. Question: The sensitivity studies (such as pressurizer location, loop nodalization and break flow nodalization studies) in Appendix A were performed using the base nodalization and a particular set of user input options. Discuss the potential for the results of the sensitivity studies to be affected by the input to the model, and justify the applicability of the results of the studies to all the plant types to be analyzed. For example, would the results of these studies be different for the different Westinghouse designs (three or four loop plants) and CE designed plants because of different input?

Response: The evaluation model is a guide to the application of a series of codes for the purpose of calculating the response of the reactor system to a hypothetical loss-of coolant accident. The model rests for its base on the sound application of the physics of the problems solved, direct experimental correlation, selected integral experiments, theory, sensitivity studies, peer review, and selected conservatisms both dictated and unilateral. For large breaks the transient is characterized by rapid, violent system depressurization that can produce large gradients in fluid conditions throughout the system. These, in turn, affect the depressurization rate (flashing), core flows and cooling, and leak flows. The sensitivity studies are performed to understand the interaction of parameters and to determine their generalized impact on the overall system responses. This understanding and the generalized interaction of parameters is broadly applicable to all light water reactor and fuel designs. Only the specific result of a sensitivity study will change from plant to plant or fuel design to fuel design.

The studies performed for the B&W recirculating steam generator plant evaluation model, provide both generalized information and specific results. As mentioned the generalized information can be directly applied to plant configurations or designs which differ from the base on which the sensitivity study was performed. To the extent that a specific result is important to a LOCA evaluation, however, that result may have to be redetermined for an alternate plant design and would certainly need to be justified.

The following discussions of each of the sensitivity studies documented in Appendix A of BAW-10168, the B&W RSG Evaluation Model topical, develop the generalized and specific results, knowledge, gained from the study and documents the extent to which the studies will be applied to alternate plant designs. For convenience, the studies are listed in the order that they appear in Appendix A and by the applicable subsection number. Three characterizations of the studies have been developed:

1) <u>Generic:</u> The study developed or verified general guidance, is broadly applicable, and will be applied to the alternate plant types covered by the evaluation model.
2) <u>Confirmable:</u> The study developed or verified general guidance that is expected to apply broadly but for which judgement is reserved pending the results of a base case. For these studies, application of the evaluation model to the alternate design will be made and parameter behavior compared to the reference sensitivity study base to confirm that sufficient uniformity of results exist to apply the study to the new design. The rationale and reasoning used to justify the application of the study to an alternate plant type will be presented in the application topical for the plant type for NRC review and approval. If the study cannot be applied or extended and the information is still required for the application of the evaluation

model, it will be repeated with the first applications report for that plant type.

3) Specific: The study developed or verified a specific result and is therefore not broadly applicable. If the study cannot be extended or justified and the information is still required for the application of the evaluation model, it will be repeated with the first applications report for the plant type.

A.2.1. RELAP5/MOD2-B&W Time Step Study

Generic. This study verified in a light water reactor geometry that the RELAP5 time step controller governs the code solution sufficiently to assure converged results. Alternate system designs within the group to be covered by the evaluation model will not change that result.

A.2.2. RELAP5/MOD2-B&W Loop Noding Study

Generic. This study verified the general noding requirements within the loop for recirculating steam generator plants. In conjunction with the break noding study the results can be applied to the separate regions of the hot leg, the steam generator, and the cold leg. Alternate system designs within the group to be covered by the evaluation model will not change the noding requirements.

A.2.3. REFLOD3B Loop Noding Study

Confirmable. This study verified the noding detail used in the REFLOD3B code. It is applicable to plants with a one to one correspondence of hot and cold legs. A separate study will be performed for the first application to the Combustion 2-by-4 design to confirm the noding detail. For other system designs within the group to be covered by the evaluation model, the results of this study will be applied.

A.2.4. REFLOD3B Primary Coolant Pump Rotor Resistance Study

Generic: This study showed a considerable reduction in flooding under a locked rotor assumption. The study affirms the generally accepted data on loop resistance effects on reflooding rates and will be applied for all system designs covered by the evaluation model.

A.2.5. REFLOD3B Maximum ECCS Injection Study

Specific. Until recently, the prevailing results have shown that the maximum ECCS case will be more severe. The study will be performed on a plant specific basis.

A.3.1. RELAP5/MOD2-B&W Break Noding Study

Generic. Per Appendix K of 10CFR50.46, for design basis LOCA's the break location is always within one of the reactor coolant system pipes or one of the attached pipes. This study verified that hydraulic stability is achieved by providing at least one control volume in the pipe between any adjacent component and the break node. The study is applicable to all plants covered by the evaluation model.

A.3.2. RELAP5/MOD2-B&W Pressurizer Location Study

Generic. Although the assumption placing the pressurizer in one of the intact loops was somewhat conservative, this study showed that there is little difference in results when the pressurizer is modelled in the broken loop. The lack of sensitivity to pressurizer location is expected to hold for all designs covered by the evaluation model and this study will be not be repeated.

A.3.3. RELAP5/MOD2-B&W Pump Degradation Study

Confirmable. This study established a most severe pump degradation multiplier by altering the pump effects on the core flow. Although it is expected that the LOCA core flow histories for all plants covered by the evaluation model will be similar, that should be established prior to adopting the results of this study. A base evaluation of any alternate design will be conducted on the first application of the evaluation model to that design. If the core flow history of the new design generally agrees with the history of the reference design, this study will be applied to the new design. On the other hand, if it does not, the study will be repeated.

A.3.4. RELAP5/MOD2-B&W Core Noding Crossflow Study

Generic. This study verified that cross flow in a light water reactor is limited and does not alter the course of a LOCA evaluation substantially. The study is dependent only on the very basic aspects of the fuel design, which are consistent across the range of designs to be considered, and will be applied for all system designs covered by the evaluation model.

A.3.5. RELAP5/MOD2-B&W Core Noding Study

Generic. In conjunction with the core cross flow study this study verified that the modelling of light water reactor core in six axial segments with a hot and an average channel provides sufficient spacial detailing for both model convergence and results accuracy. As the basic core arrangement and fuel design is not altered across the range of designs to be considered, the results of the study will be applied for all system designs covered by the evaluation model.

A.3.6. FRAP-T6-B&W Time Step Study

Generic. This study verified that the time step selection for FRAP-T6-B&W provided converged results for the spacial detail modelled in the base runs. Because, as per the discussion for the next study, the spacial detail required for the FRAP-T6-B&W model will not be altered for the other designs covered by the evaluation model, this study will remain valid and be applied for all designs.

A.3.7. FRAP-T6-B&W Radial Fuel Segmentation Study

Generic. This study verified that the number of solution points selected for radial representation of the fuel pin used by the base FRAP-T6-B&W model was adequate. The study is dependent only on the very basic aspects of the fuel design, which are consistent across the range of designs to be considered, and will be applied for all system designs covered by the evaluation model.

A.3.8. Spectrum Analysis

Specific. This study is not actually a sensitivity study from the standpoint of input selection for the evaluation model, but was included to show that the sensitivity studies performed were done with break size and location which, if not the actual worst case, were very close. A minimum of three break sizes will be analyzed on the first application of the evaluation model to any new design. If a larger spectrum is required to identify the worst case LOCA then it will be performed.

A.3.9. Time-in-Life-Study

Confirmable. This study showed that the initial fuel temperature dominated the results of variable variation across the range of burnups currently being licensed. The results will remain valid so long as the maximum burnup or the fuel design is not altered such that a high pressure fuel pin can be made to rupture during blowdown (all ruptures occurred in the post blowdown period for the study). If an examination of the fuel performance and a base run for the alternate design can-not assure that the end-of-life fuel pin conditions will not result in a blowdown rupture, a specific calculation of the condition will be performed.

A.3.10. Most Severe Break Case

Specific. This study is not a sensitivity study and was included only to show cumulative impact of the sensitivity studies and model updates performed. Any application of the evaluation model will include a unique affirmation and calculation of the most severe break case.

43. Question: With regard to the LBLOCA assessment of Semiscale Test S-04-06 using RELAP5/MOD2-B&W described in Appendix G of BAW-10164P:

a. Page G-7 stated, "The Cycle 36.04 pressure response near the broken loop simulated pump suction side, as shown in Figure G.1-6, supports the conclusion made from Figure G.1-4 that the HPI flow rate difference is the cause for the prediction of higher pressure than the data in the 1.0 to 8.0 second time period." Clarify how Figure G.1-6 supports the conclusion made regarding the pressure comparison in Figure G.1-4. Also, why were the measured and input HPI flow rates different?

Response: In response to Question 12 of first round of questions on the RELAP5/MOD2-B&W topical report, Semiscale MOD-1 Test S-04-6 was reanalyzed. Section G.1 of BAW-10164 was rewritten and submitted to the NRC. Therefore the response to this question will be based on the reanalysis results.

Figures G.1-4 and G.1-6 are replaced by Figures 12.5 and 12.7 of Question 12, and page G-7 is replaced by page 18 in response to Question 12.

The input HPI flow rates are the average values from the measured data, and these input values are the same as those used in the RELAP4 model given in Reference 7 of BAW-10164. The base case prediction of higher pressure near the pump side, shown in Figure 12.5, cannot be explained by the difference in the HPI flow rates. The difference can be due to the difference in the calculated versus measured ECCS flow rates and due to the interaction of the ECCS fluid with the system fluid near the break location. From Figures 12.5 and 12.22 it can be observed that the pressure in the EM case started deviating from the data after the accumulator injection started, which seems to indicate the effect of condensation. Therefore, the following changes to the text are made.

The sentence, " The difference between the measured and the input values of the HPI flow rates near this break location is the cause of this difference," is to read, " The difference between the measured and the calculated values of the ECCS flow rate and the interaction of the ECCS water with the system fluid near the injection location is believed to be the cause of this difference."

The sentence, "The Cycle 36.04 pressure response near the broken loop simulated pump suction side, as shown in figure 12.7, supports the conclusion made from Figure 12.5 that the HPI flow rate difference is the cause for the prediction of higher pressure than the data in the 1.0 to 8.0 second time period," is to be replaced by, "This suggests that the primary system pressure is strongly influenced by the vessel side break than the pump side break as shown in Figures 12.4, 12.5 and 12.7."

b. Page G-9 listed the differences in the core flow calculated by the base and EM cases as compared to the test data. Clarify the reason or reasons for the differences in core flow shown in Figure G.1-18.

Response: Figure G.1-18 is replaced by Figure 12.20 and page G.9 is replaced by page 21 of the answer to Question 12.

From the Figure 12.20 it can be seen that both the base case and the EM case predicted generally lower flow rates in the lower plenum (hence in the core) than the data. The effect of this lower flow rate prediction is reflected in the prediction of higher cladding temperatures as shown in Figures 12.32 and 12.33.

In Test S-04-6, the pump was coasted down from 2400 rpm to 1500 rpm which was held constant for the duration of the transient. The flow rate into the downcomer from the intact loop cold leg, shown in Figure 12.19, is a measure of the pump performance. From Figure 12.19 it can be seen that the code correctly predicted the flow rate for about 6 seconds. From 6 to 14 seconds, the code generally calculated lower flow rates than the data observed.

The vessel side break flow rate is shown in Figure 12.14. The difference between this break flow rate and the intact loop flow rate into the downcomer is provided by the downcomer. In the calculation, the downcomer was found to be providing more flow to the break than it received from the lower plenum. For example, in the EM calculation the break flow was higher than the data after about 7 seconds. During this period the flow rates into the downcomer from the intact loop (Figure 12.19) and from the lower plenum (Figure 12.20) were lower than the data. Therefore, the downcomer would be voiding slightly faster in the prediction than in the test.

During the early part of the transient the lower plenum flow rate reflected the vessel side break flow rate. From 0.0 to about 1.5 seconds the EM break flow rate was higher than the data which was reflected in the lower plenum flow rate (Figure 12.20). From about 1.5 to 2.5 seconds the EM break flow rate was lower than the data.

As a result, the calculated flow rate in the lower plenum was lower than the data during this period.

c. The calculated and measured fluid temperatures at the core inlet and in the upper plenum were compared in Figures G.1-28 and G.1-29. Although superheated vapor was calculated in both the base case and EM analyses, neither is as high as the amount of superheat in the test. Clarify the reason for this difference. Could this difference affect the ability of RELAP5/MOD2 - B&W to properly calculate temperatures at the end of blowdown in a licensing calculation?

Response: Figures G.1-28 and G.1-29 are replaced by Figures 12.30 and 12.31 of the answer to Question 12.

The important parameter from the licensing point of view is the cladding (fuel rod) temperature. The parameters that affect the core heat transfer (and hence the cladding temperature) are the core flow rate and the core fluid temperature. For a conservative cladding temperature prediction, the core heat transfer to the fluid should be lower. As explained in the previous section, as part of question 42.b, both the base and the EM models predicted lower core flow rates and higher cladding temperatures than the data during the blowdown period and therefore both predictions are conservative for licensing applications. The lower core flow rates and lower core heat transfer will affect the core fluid temperature. It is to be noted that it would be difficult to obtain the average fluid temperature condition in regions like the lower plenum and the upper plenum using one or two thermocouples. Therefore, the calculated fluid temperatures shown in figures 12.30 and 12.31 do not actually represent the local measurements.

From Figures 12.30 and 12.31 it can be observed that in the test as well as in the prediction the fluid in the lower plenum and in the upper plenum remains saturated during the major portion of the blowdown period. The differences between the measurements and the prediction near the end of blowdown can be attributed to the reasons explained above. Indeed, from Figure 12.30 it can be seen that the calculated vapor temperature in the lower plenum does show superheated steam conditions near the end of blowdown.

The effect of the superheating predicted by RELAP5/MOD2-B&W in the core region on licensing calculations is limited to the fluid temperature used by FRAP-T6 for the last one or two seconds of blowdown. The initial condition for the core in REFLOD3B is selected as saturated steam for conservatism. The density of saturated steam, being higher than that of superheated steam, allows a larger core steam mass which superheats in the loops once reflooding starts. This causes a higher degree of initial steam binding than would occur if the core was initialized in a superheated condition and slightly retards the initial flooding rates. The FRAP-T6 evaluation after blowdown is adiabatic during the refill period and uses BEACH heat transfer coefficient that are normalized to the saturation temperature with the fluid temperature assumed to be the saturation temperature after the refill period. The BEACH code is initialized with superheated steam in the core at the same temperature as the cladding. Therefore, except for the last one or two seconds of the blowdown the prediction of superheat in the core region is not used in the prediction of the cladding temperature and does not effect those results. The acceptability of

RELAP5/MOD2-B&W for licensing calculations is demonstrated by the prediction of conservative cladding temperatures as shown in Figures 12.32 and 12.33.

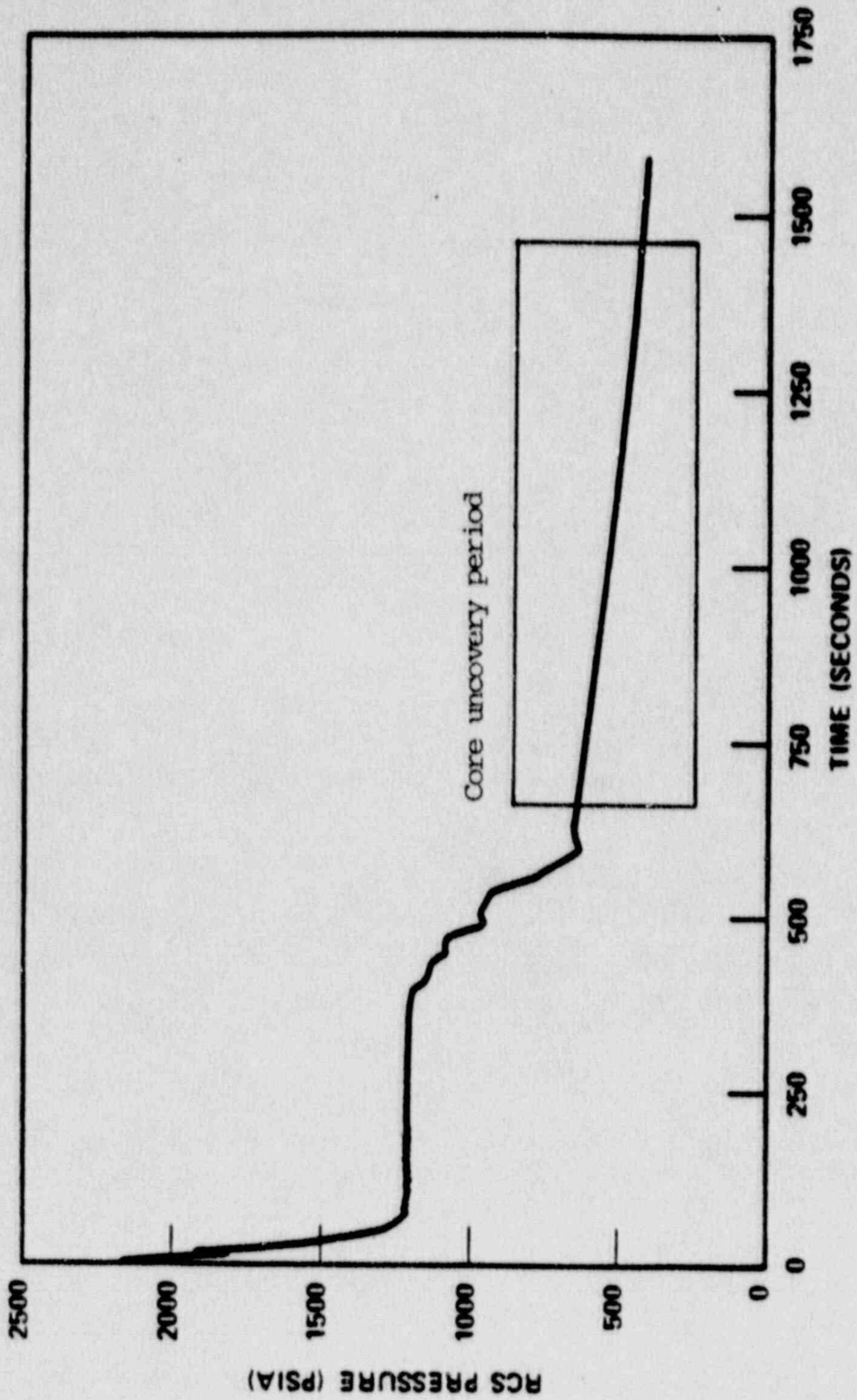
62. Question: Item I.D.2. of Section 4.4. stated that containment pressure was not a critical factor for SBLOCA's as long as the value used was reasonable. Will the value used be justified on a plant specific basis? If not, provide additional information to clarify B&W's approach to meeting the Appendix K requirement. This includes a discussion on what is meant by a "reasonable value", the containment pressure B&W intends to use, and justification that this pressure is applicable to all the plants to be analyzed.

Response: Most small break LOCAs experience only critical or choked break flow and all SBLOCAs spend most of the transient under conditions of choked flow. Figures 62-1 through 62-2 show system pressure histories for representative SBLOCA transients taken from FSARs for plants designed by Westinghouse and B&W. The timing of the occurrence of peak cladding temperature is indicated by a small box on each curve. Current PWR containment designs have LOCA pressures that range from subatmospheric tc 50 or 60 psia. As can be seen from the pressure ratio between the system and the containment the leak flow for this class of small breaks must be choked for times near the occurrence of peak cladding temperatures. Thus only the larger, generally much less severe, of the SBLOCA breaks can experience subcritical flow. So long as choked flow exists, the containment pressure is irrelevant to the course of tne accident. In fact, for events for which the peak cladding temperature occurs at a system pressure of 100 psia or more, the containment pressure is immaterial.

The larger class of small breaks may evolve at low enough pressures to interact with the containment pressure. Such accidents are mitigated by the low pressure injection system, do not have the extended periods of core uncovery (usually do not have any core uncovery), and do not involve a core reflood period. Interaction between the LOCA and the containment pressure can be conceived of in two areas. The first involves the possibility that lowering of the containment pressure might extend blowdown and encourage liquid entrainment to the level that the RCS inventory is excessively decreased. This effect takes place when system inventories are controlled by the flow and entrainment processes present in large break LOCAs and is one of the phenomena used to differentiate between the large and the small breaks. Because the effect is part of the division between the two classes of LOCA it is appropriate for consideration in the large break class but does not affect small breaks (the LBLOCA model includes special provisions and assumption on containment pressure which conservatively enhance this aspect of the transient). Therefore, for SBLOCA evaluations the containment pressure selected should not be skewed for the purpose of enhancing entrainment of liquid.

The second effect involves the relationship between the containment pressure and the ECCS charging rate. The higher the containment pressure, the lower the ECCS charging rate and vice versa. Thus, for conservative evaluations, the containment pressure should be set at or above that which is realistically expected. To assure conservatism, the containment pressure used in the SBLOCA evaluations will be held constant at or above the highest containment pressure reported in the FSAR for any event (may not even be a LOCA). This pressure is reasonable and slightly higher than expected, creating an underprediction of the low pressure injection ECCS flow and therefore a conservative evaluation of the accident.

Figure 62.1 SBLOCA PRESSURE FOR WESTINGHOUSE
DESIGN



RCS PRESSURE 4 INCH
SMALL BREAK

CATAWBA NUCLEAR STATION

Figure 62.2 SBLOCA PRESSURE FOR B&W DESIGN