ENCLOSURE 3

SECOND POUND

4.0 REACTOR

A. With regard to question 4.2.8, the response is insufficient to allow an adequate evaluation. The discussion indicates that vibration testing of operating reactor internals has verified that the vent valves do not undergo excessive vibration.

Identify which specific instrument or combination of instruments described in BAW-1003[illegible] led to this conclusion. Justify this instrument's (strain gage, accelerometer or pressure sensor) ability to detect a vibrating vent valve. Specify the expected exciting frequency of a vent valve and discuss any plans to instrument the Davis-Besse vent valves.

B. With regard to question 4.4.1, the response is insufficient to allow an adequate evaluation. Page 1-25 states, "Table 1-3 identifies all the significant changes that have been made in the station design since submittal of the Preliminary Safety Analysis Report (PSAR)." The Regulatory staff previously noted several inaccuracies in this table and requested that the table be corrected. Since Toledo Edison is apparently in conflict with the staff's interpretation of the word "significant" (vent valves, ECCS cross-overs, new main steam line rupture trips, new feedwater line rupture trips, auxiliary feedwater piping re-design, automatic switch to recirculation mode, rod worth changes, etc.), our position is that this table must now reflect <u>All</u> changes specified in question 4.4.1 since the PSAR.

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C. With regard to question 4.4.3, unless actual operating information from operating B&W plants would indicate that all vent valves are in their normally closed position, the staff's position remains unchanged; that is, one vent valve less than the minimum detect able number of stuck open vent valves shall be assumed to be open for the analyses of the thermal-hydraulic design of the reactor coolant system and core and for all transients. Either submit the re-analyses or:

1. Show that a stuck open vent valve would be detected by an operator or,

2. Show that valves are normally closed on operating reactors (wear on components, inspections, etc).

Also, the response states that the valves can be tested during each refueling. Submit (or reference) your proposed Technical Specification which adopts this surveillance requirement.

D. Revision 10 to the FSAR (page 4-85) increased the number of control rod assemblies from 49 to 53, with a resultant increase in total worth from 8.3Δ k/k to 10.0Δk/k. Explain why such a significant reactivity control modification is necessary at this time.

E. With regard to the response to question 4.2.2 (Part h), describe in detail the modification to the prototype Type-A roller nut control rod drive mechanism and show that this modification represents no unwarranted extrapolation of prototype testing technology. Why is this modification not shown in Table 1-3, "Comparison of Final and Preliminary Designs?" List the B&W reactors which incorporate the Type-C mechanism.

5.0 REACTOR COOLANT SYSTEM

A. With regard to question 5.2.4, the response is insufficient to allow an adequate evaluation. The requested discussion should be provided for pressurizer and steam generator safety valves and relief valves. Also, submit the following material:

1. A detailed description to accompany the requested diagrams explaining pressurizer safety and relief valve operation and identifying if and when credit for pressurizer electromagnetic relief valve operation (2255 psig set point) is assumed in Chapter 15.0.

2. A discussion of whether consideration of backpressure has been factored into the safety valve sizing analyses.

3. Do the pressurizer and steam generator safety valve sizing analyses assume the failure of one valve in each instance?

4. BAW-10043 is not an acceptable reference for Davis-Besse. The response states that the plant analyzed in BAW-10043 is the same size as Davis-Besse; however, the number of steam safety valves required on Davis-Besse (18 valves) does not reflect the analytical conclusions in the topical report (22 valves-BAW-10043).

Also, the steam generator design pressure is 1050 psig (FSAR page 5-68). The peak steam generator pressure for the feedwater temperature decrease transient (page 15-62) is about 1175 psia. Since this transient is more severe than the sizing transient, BAW-10043 could not be applicable. We also note that the pressure in this case reaches, and may even exceed, the 110% design criterion. It is therefore the staff position that a complete pressurizer and steam generator safety valve sizing analysis must be submitted specifically for Davis-Besse. Also, modifications to the secondary system design (overpressure protection) appear warranted.

5. Compare the margins (minimum psi below ASME limit) calculated to occur in the Davis-Besse pressurizer and steam generator safety valve sizing analyses with Oconee, Rancho Seco, North Anna, and Bellefonte.

B. With regard to question 5.2.5, the answer is insufficient to allow an adequate evaluation. Using the Line Designation Table on Figure 9-17 (referenced in the response), all reactor coolant lines fail to meet minimum design requirements (see Figures 5-3 and 6-17). Resolve this concern and include design temperature as a part of the Line Designation Table.

C. With regard to question 5.5.3, the response is insufficient to allow an adequate evaluation. It is stated that during normal operation, the ECCS lines are filled with fluid but are in a no-flow condition. Describe the system provided to maintain water in the ECCS lines in spite of any expected leakage back through the provided check valves. State this system's safety design basis and provide detailed design specifications including line sizes and capacities. Our position is that the Davis-Besse design must reflect consideration of a water hammer being generated when coolant discharges into an empty line. Also describe the design features that are provided (venting, etc.) to prevent air entrapment within ECCS pump casings from reducing ECCS pump performance.

D. Recently, an unusual event occurred on Oconee Unit 3 in which the anti-rotational device on one of the reactor coolant pumps failed to function, and the pump rotated in the reverse direction. With regard to this occurrence, discuss the consequences that such a failure of the anti-rotational device could have on normal operation, transients, and accidents.

E. With regard to the response to question 9.2.5, provide a plot of reactor coolant temperature versus time (from full power operation) using one and two RHR trains. Describe and justify the procedure for attaining a cold shutdown condition with a malfunction (fail closed) of isolation valve DH-11 or DH-12 (inside the containment vessel).

6.0 ENGINEERED SAFET FEATURES

A. With regard to question 6.3.5, include on the same diagram all other protection sequences required to mitigate the consequences of this event. That is, in addition to CORE COOLING, the diagram should also depict the systems required to produce other safety actions, such as REACTOR TRIP, CONTAINMENT ISOLATION, AND PRESSURE RELIEF.

With regard to Figure 6.3.5-1 (Revision 12);

1. Why are the core flood tanks shown to be required for small breaks of approximately 0.04 ft^2?

2. Show the required sequence for breaks between 0.04 ft^2 – 0.1 ft^2, 0.3 ft^2 - 0.5 ft^2 and 0.75 inch - 0.0 4 ft^2.

3. The figure shows a requirement for low pressure injection during the short term of a CFT line break. Since your analyses show that low pressure injection is not required, this sequence should be corrected.

4. The SFAS channels should be included.

5. Why was the AFS deleted during the long term?

B. With regard to question 6.3.6, the answer is insufficient to allow an adequate evaluation. Identify the specific systems or components which exist to provide the listed services. Include any support subsystems essential to the operation of each auxiliary system or components.

The core flood tanks are shown not to require any auxiliary systems for their operation, yet the nitrogen supply is obviously essential to the operation of these tanks. Include such support systems in this list.

C. With regard to question 6.3.10, the response is insufficient to allow an adequate evaluation. Provide chronological lists of <u>all manual actions</u> that are required by the operators following a LOCA until stable long-term cooling is achieved. These lists should include both large and small LOCA, and should indicate:

1. the actual physical action taken by the operator (i.e., switch thrown, gauge checked, button pushed, etc., and the operator's physical location necessary to perform the required action).

2. effect on the reactor systems of the action (i.e., the system or items of equipment turned on, turned off, or whose operating state is changed--power source changed, water source changed, water destination changed, etc.).

3. the information required by the operator to know when or if he should perform the operation, that is, what parameter must reach what level before the operation is required, and through what instrument does the operator obtain that information, where is that instrument's readout physically located, and how is the information conveyed to the operator (meter or graph position, audible or visible alarm, etc.)?

4. the time delay in each case during which his failure to act properly will have no unsafe consequences, and the consequences if the action is not performed at all.

Also, include all of the information requested above for automatic operations which are to be verified by the operator, indicating what manual actions he is required to take if the automatic action is not properly executed.

In regard to the specific manual actions cited in the response to question 6.3.10, it is not clear why such an operational procedure is required. All safety analyses assume a single failure of a complete LPI chain, thereby ensuring that such loss of LPI flow as is referred to in the answer to question 6.3.10 does not adversely affect the health and safety of the public. Justify the need to require these two operators to perform these potential high-radiation duties in all cases that the operator notes less than 1500 gpm in one of the DH injection lines. Also, the following points are noted:

1. Two operators are required to effect the safety action (LPI flow equalization).

2. The performance of the safety action depends on the amount of radiation in the area. No radiation levels are discussed and it is clear that this decision (no safety action versus high radiation exposure) should not be imposed on the operators.

3. No times are given.

4. No consequences are discussed.

5. No statement is made as to whether or not the need for this procedure was sensitive to break size.

6. No rationale is discussed as to why there would be less than 1500 gpm in the low pressure injection line.

7. It is not obvious that sufficient personnel are left in the control room after these two operators leave to respond to the procedures required during an accident (monitoring plant parameters, handling communications, etc.).

D. With regard to question 6.3.13, the response is insufficient to allow an adequate evaluation. The statement is made that the maximum calculated control rod centerline temperature is 1295°F at 43 seconds and that the melting temperature is 1470°F. Our concern is that temperatures may be high enough within the control rods to compromise rod integrity. The following additional information is required:

1. Confirm that the 1295°F was the maximum predicted temperature throughout the control rod.

2. What cold leg break was assumed for the analysis (split or guillotine)?

3. FSAR Figure 6-38 shows the peak fuel clad temperature occurring at about 33 seconds. Account for the 10 second difference between the peak centerline fuel temperature and the peak centerline control rod temperature.

4. Overlay a plot of maximum control rod temperature onto a plot of peak fuel cladding temperature. Identify the location of these peaks in the core. Show the plot through the reflood peaks.

5. Provide the data base for the poison material melting point.

6. Confirm that the analysis was conducted using 102% of 2772 MWT.

7. Discuss analytical methods and describe all calculations.

8. Discuss the uncertainties associated with this calculation.

9. If calculated temperature are high enough, discuss all aspects of the consequences of molten poison material within the stainless steel clad. Address such items as phase changes and degradation of the control aspect of poison material, pressure buildup within control rods; flow blockage potential due to bowed, ballooned or collapsed control rods, and all local and gross core effects.

10. For the preceding types of control rods, assess the potential for eutectic formation between dissimilar metals.

E. With regard to question 6.3.14, the response is incomplete. What is the melting point of the three combinations of B_4C-Al_2O_3?

F. With regard to the response to question 6.3.18, explain how the apparent analytical error resulted in a completely different worst break size and worst break location.

G. Provide the analytical basis for all pressure setpoints associated with the Core Flooding Tanks (600 psi for CFT actuation, 700 psi for valve position alarms, 800 psi for valve interlocks).

H. With regard to question 6.3.18 (and in light of your clarification of the FSAR statement that a change of position of valves is considered incredible applied only to the core flood line isolation valves), confirm that the consequences of such a single failure as a valve change-of-state (simultaneous with an accident) was considered plant-wide.

I. With regard to question 6.3.21, the Regulatory staff's positions that the proposed LPI-to-HPI crossover and LPI-to-LPI crossover are not acceptable designs remain unchanged. The Davis-Besse design must be modified as requested, the FSAR must be revised to reflect these design modifications, and the information requested in question 6.3.21 (part b) must be provided.

Also, the Regulatory staff notes that a break in the CFT line when the assumed diesel failure is the one upon which the normally closed injection valve in the intact CFT line is dependent (valve fails to open) renders the LPI crossovers ineffective. The design modifications adopted above must also reflect consideration of this situation. Relying on the operator to repair the failure is not acceptable.

J. Revision 9 to Figure 6-17 now shows previously closed low pressure injection valves DHIA and DHIB to be open. Why?

1. It is the Regulatory staff's position that the over-pressure protection now in these ECCS lines is insufficient. The number and type of valves used to form the interface between the low pressure ECCS discharge and the reactor coolant system must provide adequate assurance that the ECCS will not be subjected to a pressure greater than its design pressure. This may be accomplished by any of the following methods:

a. One or more check valves in series with a normally closed motor operated valve. The motor operated valve is to be opened upon request of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

b. Three check valves in series.

c. Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

2. Why are check valves DH 76 and DH 77 shown to be locked open (see Figure 6-17)?

K. Describe the sequence of events which take place during the automatic switch to the long-term recirculation cooling mode of operation. Provide a system description and state the rationale for the relative order in which these automatic actions occur. Describe any operator action required to shift into and maintain this mode.

L. Regulatory Guide 1.79 provides additional guidance for pre-operational testing of ECCS. Evaluate the planned Davis-Besse testing program with this Guide and itemize the areas of non-conformance. Toledo Edison will be required to conduct a test under ambient conditions that demonstrates the capability of the system to operate in the recirculation mode of ECCS operation. Our specific concerns are the possibility of inadequate NPSH, air bindage, or vortex formation at the sump screens, any of which could adversely affect ECCS performance. Discuss how your proposed test program will address these concerns. To avoid reactor coolant system contamination, the sump water may be discharged to external drains or other systems. Temporary arrangements may be made to provide adequate sump capacity for pump operation.

M. A recent occurrence at Oconee allowed several feet of water to build up in an ECCS pump room and consequently jeopardized the availability of ECCS pumps. Evaluate and provide means for decreasing the potential for a similar occurrence in Davis-Besse. Does Davis-Besse have sump pump monitor alarms? Are the ECCS pump rooms watertight?

N. In a recent occurrence at Oconee, cavitation damage to ECCS valves was reported due to decay heat removal operation at certain flow rates. Evaluate and provide means for eliminating the cavitation problems associated with flow rates through these valves in Davis-Besse.

O. The staff has noted that Davis-Besse Figure 6-17 shows TWO High Pressure Injection Pumps (550 gpm each), while THREE (of varying capacities from 250 gpm to 700 gpm) are shown on such plants as Rancho Seco, Oconee, WPPSS-1, Greenwood, and Bellefonte. Explain this variance in HPI design concept on Davis-Besse.

15.0 ACCIDENT ANALYSES

A. With regard to question 15.1.2 (partial loop operation), the response is partially acceptable. Provide the following additional information:

1. A plot of peak clad temperature versus time for the worst case LOCA during two-pump operation.

2. For two-pump operations, evaluate the applicability of each generic sensitivity study presented in topical report BAW-10091.

3. For the Uncompensated Operating Reactivity Changes event, why were the assumed maximum reactivity rates the same for 2-and 3-pump operation, yet quite different from the rates for 4-pump operation on page 15-66 of the FSAR? Why did the rate of average RCS temperature change for xenon buildup become less negative in going from 2-pump operation to 3-pump operation, yet more negative in going from 3-pump operation to 4-pump operation? Similarly, why did the rate of average RCS temperature change for xenon burnout decrease in going from 2-pump operation to 3-pump operation, yet the rate increased in going from 3-pump operation to 4-pump operation?

4. For the worst-case main steam line break, provide a plot of Total Reactivity (% Δ k/k) versus Time After Break (seconds) for 2-pump operation.

5. Provide a plot of DNBR versus time for the rotor seizure during 2-pump operation. Justify the initial DNBR.

B. With regard to the control rod assembly group withdrawal event presented in subsection 15.2.2:

1. Provide a plot of maximum Reactor Coolant System Pressure during the transient versus Initial Power Level (similar to Figures 15.2.2-6 and 15.2.2-7). Plots of these parameters for both the Single CRA Group Withdrawal event and the All CRA Groups Withdrawal event should be included.

2. In comparing the maximum reactivity addition rates between the startup event and the rated power event, explain why there is a variation in assumed single group CRA worth and single group reactivity addition rate, especially since such SAR's as B-SAR-241 and Greenwood appear to indicate that these parameters do not change between the two events.

C. With regard to the locked rotor event presented in sub-section 15.2.5:

1. Criterion 2 on page 15-37 states that no fuel cladding failure shall occur. Page15-2 states:

   "The criterion, adopted in these accident analyses to ensure that no fuel damage occurs, is that a DNBR greater than 1.3 must be maintained throughout the transient."

Since it appears that the DNBR consequences of this event violates the Babcock and Wilcox criterion for acceptability (DNBR = 1.05, Table 15.2.5-3), explain this inconsistency.

2. Page 15-2 states:

   "If the DNBR goes below 1.3 during a transient, the gap activity for all of the fuel rods with a DNBR of less than 1.3 is assumed to be released."

   This assumption appears to have been overlooked in the locked rotor event since page 15-39 states that, "No fission product release is postulated for the locked rotor event." Confirm that the gap activity for all of the fuel rods with a DNBR of less than 1.3 was assumed to be released to the reactor coolant. Provide the percent fuel rods involved in this release.

3. Discuss the conservatism in the initial DNBR shown in Figure 15.2.5-5 (about 1.9) and compare the initial value assumed for the 4-pump coastdown event. Justify any difference in assumed values.

D. With regard to question 15.2.12 (feedwater line break), the response is insufficient to allow an adequate evaluation.

1. For the feedwater line break <u>with offsite power available</u> (Case I):

   a. Justify the effective area (0.5 ft^2) of the break in an 18-inch main feedwater line.

   b. Reanalyze the event at 102% of rated power (2% to account for power uncertainty).

   c. Justify the closure times of the feedwater stop valves and provide the expected ΔP against which these valves would be closing to achieve these isolation times. Provide any available test data to verify this capability.

   d. What is the effect upon the flow rate of the line losses associated with the recent addition of the auxiliary feedwater crossover lines. Submit calculations.

e. Justify the quality of two-phase mixture exiting the steam generator via the break and discuss why the blowdown could not be initially single phase (feedwater). Relate this same discussion to breaks at other elevations on the steam generator (e.g., auxiliary feedwater line).

f. Page 15.2.12-2 states that the reactor trips on high reactor system pressure (2355 psig setpoint from page 2.2-1 of Chapter 16.0). Since the pressurizer electromagnetic relief valve setpoint is 2255 psig (response to question 5.2.3), analyze the consequences and extent of delay of the pressure trip due to the relieving capacity of this valve (and therefore more time for energy to build up in the core before reactor trip). State the capacity of this relief valve and provide the tolerances in ± psig. Include the uncertainty in ± psig of the reactor trip pressure setpoint.

Also, include a discussion of the potential delay in reactor trip due to the probable actuation of the pressurizer sprays.

g. The explanation on page 15.2.12-2 which compares the describe break situation with the event analyzed in the FSAR is not detailed enough to be clear. For example, the statement is made that for the described accident, the reduction in the secondary system heat removal capability..." is not extreme since only the unaffected steam generator heat removal capacity is reduced." Doesn't the complete loss of the affected steam generator reduce heat removal capacity?

This discussion also states that, "Since the results of both accidents are similar, Figure 15.2.8-1 is sufficient to show the eventual effect of a feedwater line rupture upstream of the first feedwater line upstream check valve with offsite power available." This statement makes little sense because it conflicts with page 15.2.12-1 of the response which placed the break downstream of the check valve (in accordance with the staff question). Also, the FSAR (page 15-53) states:

"The loss of normal feedwater due to a feedwater line break between the first feedwater line up-stream check valve and the steam generator produces results no worse than the steam line break accident presented in Section 15.4.4."

This above conclusion from the submitted information is not obvious, especially after noting the following:

- The reactor trips on low pressure for the main steam line break, but on high pressure for the feedwater line break.

- The critical time period in the main steam line break shows a reactor coolant temperature and pressure decrease, but the critical period in the feedwater break shows a coolant temperature and pressure increase.

2. With regard to the feedwater line break with a loss of offsite power concurrent with the feedwater line rupture (Case II):

   a. The statement that the event is less severe than, but similar to, the results of the station blackout (in lieu of the requested analyses) is not acceptable.

   b. Amend the FSAR to verify that the plant auxiliary boilers are not used to supply steam to drive the auxiliary feedwater system.

   c. Discuss the term "high level control" which is stated to be initiated by reactor trip on the secondary side of the steam generators to enhance natural circulation. Provide a description and available data to substantiate and quantify this operating capability. Subsection 5.5-2, "Steam Generators," does not appear to address this safety feature. Is it required to mitigate the consequences of the feedwater line break?

   d. Page 15.2.12-3 of the response indicates that the unaffected steam generator pressure increases to the turbine bypass valve setpoint and the steam line safety valve setpoint to remove decay heat.

   However, page 15-57 of the FSAR states that for a loss of offsite power..."turbine bypass valve steam relief is lost due to the loss of power to the condenser circulation pumps." Please explain this apparent conflict and, if credit for turbine bypass was improperly utilized for the feedwater line break, re-assess the consequences of this event.

3. With regard to the break analyzed with a loss of offsite power at reactor trip (Case III), the brief discussion provided in inadequate to quantify any consequences and leaves unanswered the concern as to whether or not this is a worst-case situation relative to the status of offsite power.

E. The response to question 15.2.13 is partially acceptable. With regard to the overpressure protection of the RHR system, provide a discussion of the design features which protect the RHR system against overpressurization during shutdown (while the shutdown cooling system is functioning). Include detailed analyses, with calculations, of needed RHR relief valve capacity, if applicable. Justify all worst case events considered in the relief valve sizing analysis (see Chapter 15.0 events). Include consideration of starting up with a "solid" pressurizer at the time of the pressure transients. Describe, with diagrams, the relief valve design and operation.

F. With regard to question 15.2.14 (loss of offsite power), the Regulatory staff review of the computer code POWER TRAIN used for this event (BAW-10070) is not complete. Should modifications to this code be required, the impact of such modifications upon the consequences of this event will have to be assessed.

1. The top of page 15-58 states:

   "Excess steam is relieved until the reactor coolant system pressure is below the pressure corresponding to the lowest setpoint of the steam safety valves."

   With regard to this statement, state the setpoints, and capacity at these setpoints, of the steam relief and safety valves. How is the reactor coolant pressure incorporated in the operating logic of the secondary system safety valves (as indicated by the statement)? Describe the basis for this arrangement.

2. Page 15-58 of the FSAR also states:

   "The turbine-driven auxiliary feed pumps provide feedwater to the steam generator by taking suction from the condensate storage tanks and are driven by steam from either steam generator."

   With regard to this statement, it appears to conflict with the response to question 15.2.12 which indicates that for a feedwater line break with a loss of offsite power the operator would have to rely upon the plant auxiliary boilers to supply steam to drive the auxilairy feedwater pumps. Please clarify.

3. Comment on the validity of the statement in the introduction to Chapter 15.0 (page 15-2) that all systems utilized in these Chapter 15.0 analyses have been designed in a manner such that a single failure of an active component will not prevent then from meeting their performance requirements.

G. With regard to question 15.2.15 (feedwater system malfunctions), the response is insufficient to allow an adequate evaluation. Address the following comments:

1. Page 15-59 states, "Normally operator or ICS action would correct feedwater system malfunctions, however, such actions were not considered in the analysis of this accident."

   This statement conflicts with the response which indicates that although credit for ICS was not assumed, the operator was required to control steam generator water level. Correct this apparent inconsistency.

   If operator action is not required, provide the analytical consequences of this event using the protective systems referred to in the following statement (page 15-59 of the FSAR):

   "Only the low reactor coolant pressure and high neutron flux trip were used in the analysis to assure reactor protection."

   If the existing FSAR plots are intented to represent this situation, then the description on page 15-60 should be clarified; specifically, the following statements are confusing if no credit for ICS or operator action is assumed:

   "Without temperature compensation of the feedwater flow the steam generator level rises to the high level limits where feedwater flow is reduced to prevent flooding of the steam generator."

   And for the feedwater flow malfunction:

   "The steam generator level rises to the high level limit where feedwater flow is reduced."

   To repeat for clarification, how is feedwater flow reduced for the analysis which states than no ICS or operator action was assumed?

2. Provide the following additional parameters versus time for the feedwater malfunctions. The analyses should be performed at 102% power (2% to account for power uncertainties):

   - Minimum DNBR

   - Pressurizer water level

   - Main feedwater flow rates

   - Turbine bypass flow rate

   - Steam generator water levels

   - Safety and relief valve flow rates (primary and secondary system)

   Identify all trip setpoints on the plots.

H. The response to question 15.2.16 is partially acceptable. Provide the specific analysis, and properly classify in the FSAR, the inadvertant opening of a pressurizer safety or relief valve (highest capacity). State the acceptance criteria, the computer model utilized, and all initial conditions and assumptions. Provide the parameters versus indicated in part 1 of question K.

I. In the response to question 15.2.17, the statement is made that:

<u>...it is a routine safety analysis assumption that unless an action is guaranteed by the protection system it does not occur.</u> Therefore, no credit is taken for runbacks, interlocks, etc."

With regard to this statement, define the word "guarantee" in terms of such qualifying features as seismic and redundancy design characteristics.

J. With regard to question 15.4.1 (the steam line break), the response is insufficient to allow an adequate evaluation:

1. The analysis assumes credit for the shutdown boron addition of the HPI System. Credit for this additional shutdown margin from the HPI System is not acceptable unless it can be shown that the assumed portion of shutdown reactivity contributed by the HPI System would be injected at the times assumed. Along these lines:

    a. Justify the conservatism of the 25 second ECCS delay, noting that the ECCS delay assumed for the recent generic LOCA analyses (BAW-10091) was 35 seconds. Discuss also the breakdown of this time from the beginning of the event (time zero) to actual injection of the HP boron into the core (setpoint delay, pump start delay, transport delay, etc.).

    b. Show that the HPI actuation setpoint, high containment pressure (4 psig) or low reactor coolant pressure (1600 psig), is reached for breaks inside or outside the containment and specify the time to reach these setpoints. Choose conservative assumptions, such as maximum heat sinks to delay the containment pressure rise to 4 psig (for a break inside containment). For breaks outside containment, the time to the low reactor coolant pressure HPI initiation time should also be conservatively accounted for in the analysis.

2. Provide the minimum reactivity margin for the following five main steam line break situations. Specify the worst single active component failure for each case:

    Case I - 102% power. Break is inside containment (36" line). No offsite power.

    Case II - 102% power. Break is inside containment (36" line). Offsite power is available.

    Case III - 102% power. Break is outside containment and upstream of isolation valves. No offsite power.

    Case IV - 102% power. Break is outside containment and downstream of isolation valves. Offsite power is availab

    Case V - Hot standby or low power operation. Choose other assumptions based on worst-case above.

3. Page 15.4.1-2 of the response indicates that, for the steam break situation analyzed, the reactor trips on low pressure at 1.13 seconds after the rupture. This appears to be a significantly faster time-to-trip than other more recent B&W reactors. Provide a breakdown of th time, including all delays.

4. The analyses indicate that, depending on break size, the reactor will trip on either high-flux or low-reactor pressure. Provide a graph of time to reach high flux trip point versus break size, and time to reach low pressure trip point versus break size (both on the same plot).

5. We note that additional protection system trips have recently been added to mitigate the consequences of the steam line break (Main Steam Line Rupture Control System). The statement that the original FSAR analysis would be more severe (in lieu of a new analysis) is not acceptable. A new worst-case analysis should reflect the current design of Davis-Besse.

6. With regard to part "e" of question 15.4.1 the requested definition in terms of line size should be provided (e.g., at what point does a "minor" steam line break become a "major" steam line break?).

7. The response to question 15.4.1 states that periodic full-closure testing of the turbine stop valves will disclose any sticking conditions, so that a shut down could be made to make the necessary correction. Provide or reference this surveillance requirement in Chapter 16.0.

8. Clarify the need to include the feedwater control valves and stop valves in the Main Steam Line Rupture Control System. Are both these valves safety grade? State their required closure times and the bases for these times.

9. Page 7-52a describes the Main Steam Line Rupture Control System. Why is the trip of the turbine stop valves not included in this discussion? Where is the need and specifications for this trip addressed in the FSAR?

10. Page 15.4.1-4 of the answer to question 15.4.1 discusses and quotes pressure drops across a main steam line isolation valve. However, the question was directed at the <u>non-return check valves</u> during <u>accident</u> conditions. Provide the design capability of the

non-return check valves, as requested.

K. Provide the information listed below (1-5) for each of the following transients or accidents.

a. The worst case feedwater line break (See question D)
b. The worst case main steam line break (See question J)
c. Loss of offsite power (See question F)

1. Plots of the following parameters versus time:

-- Steam generator pressure (affected and unaffected)

-- Minimum DNBR (W-3 correlation)

-- Break flow rate

-- Safety and relief valve flow rates (primary and secondary system)

-- Mass and energy transfer within the containment (for breaks inside containment)

-- Turbine bypass valve flow rates

-- ECCS flow rates

-- Reactor coolant pressure and temperature

-- Thermal and neutron power

Carry the preceding parameters out to such a time period as to ensure that reactor conditions have sufficiently stabilized or abated. Identify all trip setpoints on the plots.

2. Credit for operator action to mitigate the consequences of these events is not acceptable unless analyses show that sufficient time exists for the operator to recognize the initiating event, ascertain the proper response, and perform the appropriate manual action. In this regard, identify the exact manual operations required by the operator to meet the acceptance criteria stated in the FSAR and bring the plant to a final, stabilized condition. Specify: (1) the information available to the operator, (2) the time delay during which his failure to act properly will have no unsafe consequences, and (3) the consequences if the action is not performed at all.

3. Substantiate the assumed worst single failure by a sensitivity study. Examples include the auxiliary feedwater steam admission valve or the turbine bypass system (bypass system actuation logic not single failure proof-page 10-18).

4. *a.* Provide a table of key initial assumptions employed in each analysis. Include in the table such parameters as initial power level, core flow, coolant system pressure at the core outlet, core inlet fluid temperature, volume average fuel temperature, and secondary system conditions (such as feedwater flow, steam flow, and steam generator pressure and temperature).

   b. Indicate the appropriateness of these values by comparing to the expected operating <u>ranges</u> for Davis-Besse (for example, operating pressure (psig) = 2185 $\pm$ xx).

5. To complement the FSAR discussions, provide for each of these three events a summary of a functional analysis of systems required. The summary should be shown in the form of simple block diagrams beginning with the event and branching out to the various possible sequences. When complete, the diagram should clearly identify each system required to function during any plant operating state. See the response to question 6.3.5 and subsequent staff comments.

L. With regard to question 15.4.2, the response to part "b" is insufficient to allow an adequate evaluation. The question refers to the statement on page 15-118 of the FSAR that,

"An additional source of fission product leakage during the maximum hypothetical accident can occur from leakage of the engineered safety features external to the containment vessel during the recirculation phase for long-term core cooling."

The response refers to Tables 6-15 and 6-16 which do not appear applicable. Provide an interpretation of these Tables with regard to leakage locations, flow rates, and leakage detection instrumentation. Include a description of any operator actions that are required with the time needed for the action.

M. Provide a list of all the transients and accidents considered in establishing the auxiliary feedwater system flow requirements and response times. For each of these transients and accidents, state the maximum delay in the initiation of auxiliary feedwater flow that can be tolerated, whether the initiation is automatic or requires operator action, and the minimum required auxiliary feedwater flow rate required to mitigate the consequences of the transient or accident.

N. What initiates the auxiliary feedwater system for a main steam line break and a single failure of the auxiliary feedwater pump on the unaffected steam generator (e.g., what automatic signal would open the <u>correct</u> crossover line between the two auxiliary feedwater system)

O. For each accident and transient analysed in Chapter 15.0 of the SAR which results in a reactor trip, the control rod worth curve (reactivity versus time) used in the analysis must be provided. Show that the control rod worth curve is based on an axial power profile and peaking factors selected to produce the worst consequences from the events analyzed. Relate the axial power limits and control rod limits specified in the Technical Specifications. Reference the Technical Specifications which require the operator to maintain this flux shape over the life of the core and the maximum allowed variation permitted in this flux shape.

P. To assess the potential severity of a steam line break inside of containment that results in both steam generators blowing down, provide the results of an analysis that assumes the single active failure that results in the most severe consequences regarding core thermal limits. For example, a single failure that results in the opening of one or more of the steam dump valves in the steam generator not supplying the broken steam line.

Clearly state all assumptions used in the analyses, including the time in core life and boron concentration in the reactor coolant. Justify the selection of the single failure assumed in your analyses.

Present plots of the following parameters as a function of time:

1. Neutron power level

2. Minimum DNBR (W-3 correlation)

3. Total Reactivity

4. Average core moderator temperature

5. Reactor coolant system pressure

6. Water level in the pressurizer

7. For each steam generator:

   a. reactor coolant outlet temperature

   b. steam pressure

   c. feedwater flow rate

8. Heat flux (average and maximum)

9. Fuel temperature (average and maximum)

Q. What instrumentation would be relied on to single out a steam generator tube failure as the cause of an event so that the reactor operator would know that the required action at 20 minutes must be accomplished? Our concern is that other possible events, e.g. a small pipe break LOCA for which no operator action is required, would be incorrectly diagnosed by the operator. The operator could then fail to achieve the proper manual action at 20 minutes. Why does the initial discharge out the break (Table 15.4.2-1; 435 gpm) differ from WPPSS, even though steam generator tube diameters appear identical?

R. Does the pressurizer go solid for any overpressure transients? If so, provide the bases for the water discharge rates through the safety valves.

S. Because of allowable operating ranges and instrument uncertainties, transient and accident analyses should not be started from nominal conditions but from offset conditions. Provide a table listing initial values for power, core flow, primary and secondary pressure, inlet temperature, DNBR, and reactivity coefficients. Include nominal values of parameters, uncertainty on each and the values used in Chapter 15.0. Justify the values of uncertainties used.

T. It is noted that the number of rods expected to experience DNB for the control rod ejection accident increased from less than 6% quoted in the PSAR (Figure 14-28) to 45% in the FSAR (Figure 15.4.3-9). Why?

Also, please explain the convex nature of the FSAR curve when comparing to the concave trend of the PSAR curve for Davis-Besse, Bellefonte, and WPPSS.

U. Provide a <u>table</u> of reactivity coefficients assumed for each of the Chapter 15.0 events. Include the time in core life (BOL versus EOL) at which each event was postulated. Note the expected ranges of the coefficients.

V. Reconcile the difference in Auxiliary Feedwater System response time on page 7-33 (60 seconds) and in the response to question 7.4.1 (40 seconds). Which response time was utilized in the analyses in Chapter 15.0?

16.0 TECHNICAL SPECIFICATIONS

A. Davis-Besse Chapter 16.0 should be updated to reflect the requirements of the latest approved set of Technical Specifications for a Babcock & Wilcox reactor. Areas needing changes include:

1. Specification 2.1 - Must consider the effects of fuel densification.

2. Specification 3.1.1.4 and 3.3.1.2 - Core flooding tank isolation valves must be locked open.

3. Low power physics testing restrictions must be included.

4. Specification 3.3 - Must reflect a re-analysis of ECCS performance to 10 CFR 50, Appendix K.

5. Specification 3.6.2 - Must include the LOCA limit curve and discussion.

B. Confirm that Davis-Besse will not allow single loop operation at any time (including testing) or provide the Technical Specifications establishing the testing conditions during which single loop operation is permitted.

C. The allowed power levels for partial loop operation in Figure 2.1-3 (3-pump and 2-pump) do not agree with FSAR Figures 5-10 and 5-11.

D. The relieving capacity of each pressurizer code safety valve (page 3.1-2) does not agree with FSAR page 5-14a (300,000 lb/hr versus 336,000 lb/hr).

E. Specification 3.3.1.2(a) states that, "Two core flooding tanks each containing 1040 ± 30 ft^3 of borated water at 600 ± 25 psig shall be available." Confirm that values of 1010 ft^3 and 575 psig were assumed in LOCA analyses to justify the operating ranges.