OCT 30 1981

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
Commissioner Roberts

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: STAFF REVIEW OF ORNL REPORT ON PRESSURIZED THERMAL SHOCK

In my memorandum to the Commission on October 9, 1981, I transmitted a copy of the draft preliminary report on pressurized thermal shock prepared by Oak Ridge National Laboratory. The staff has completed its review of the report, and the staff's evaluation is enclosed for your information. Also enclosed is a letter from Duke Power Company presenting results of their evaluation of the ORNL report.

A principal conclusion from the ORNL analysis is that an overcooling transient similar to the most severe transient that has occurred (the Rancho Seco event of March 20, 1978) will not pose a threat to the Oconee-1 pressure vessel for several more years. ORNL examined overcooling transients more severe than the Rancho Seco event (with lower occurrence probabilities) and, using fracture mechanics calculations thought by the staff to be conservative, predicted failure of the Oconee-1 vessel if the most severe transients were to occur today.

The staff has previously concluded and discussed with the Commission that the probability of occurrence of severe pressurized overcooling transients is sufficiently low that immediate corrective action is not warranted but that corrective actions may be required for some plants within a year.

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In the staff's judgment the ORNL report does not present any significant new information that would change that conclusion.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosures:
1. Staff's Assessment of ORNL's Draft Preliminary Report on Pressurized Thermal Shock
2. Letter from A. C. Thies, Duke Power Company, to R. M. Bernero, NRC, dated October 20, 1981

# NRC STAFF ASSESSMENT OF THE DRAFT INTERIM REPORT BY OAK RIDGE NATIONAL LABORATORY ENTITLED, "EVALUATION OF THE THREAT TO PWR VESSEL INTEGRITY POSED BY PRESSURIZED THERMAL SHOCK EVENTS"

## I. Purpose

The purpose of this document is to assess the regulatory implications of the draft Oak Ridge National Laboratory Report, NUREG/CR-2083, entitled, "Evaluation of the Threat to PWR Vessel Integrity Posed by Pressurized Thermal Shock Events" (ORNL report) with respect to the acceptability of continued reactor operation considering the pressurized thermal shock issue. For that purpose, the NRC staff has made a preliminary review of the report, and the staff's conclusions and supporting information are presented here.

## II. Summary and Conclusions

The NRC staff finds the ORNL report to be a useful summary of the background and present status of the pressurized thermal shock issue. As the ORNL report states, results cited in the report are drawn from previous work and literature sources.

A principal conclusion of the ORNL report is that an overcooling event similar to the most severe transient that has occurred (the Rancho Seco event of March 20, 1978) will not pose a threat to the Oconee-1 pressure vessel for several more years.

Another important conclusion of the ORNL report is that if certain events more severe than the Rancho Seco overcooling event were to occur today, and if the reactor pressure vessel were to remain at high pressure or be repressurized, and if fracture mechanics calculations believed to be conservative are used, then vessel failure may be predicted for the Oconee-1 vessel. The NRC staff had previously reached this same conclusion (Refs. 1, 2, 3, 4, 5), which is the reason that the pressurized thermal shock issue is under consideration today.

The NRC staff also had previously concluded, and discussed with the Commission, that the probability of occurrence of pressurized overcooling events, more severe than the Rancho Seco event, is sufficiently low that immediate corrective action is not warranted (Ref. 1), although longer term corrective actions may be required for some plants within a year. The ORNL report does not present any significant new information that would change this conclusion by the staff.

The report presents the completed results of analyses for four overcooling transients postulated for Oconee-1. These are: a large break loss-of-coolant accident (LOCA), a main steam line break (MSLB), an overcooling event which actually occurred on March 20, 1978 at Rancho Seco, and a postulated overcooling event more severe than the Rancho Seco event, referred to as the runaway feedwater transient (RFT). In addition, ORNL reviewed partial calculations for a small break LOCA, but the calculations were not completed for the ORNL report.

Table 8.7 of the ORNL report presents results showing effective full power years before predicted Oconee-1 vessel failure. For the large break LOCA and the Rancho Seco transient, previous results are confirmed that many years of operation remain before these events would present a potential for failure of the pressure vessel. Time-to-predicted vessel failure results for the small break LOCA are not presented in the ORNL report, but other calculations have been made which indicate this event is not of immediate concern (Refs. 5 and 6) and it will not be further discussed here.

The time-to-predicted vessel failure results presented in Table 8.7 of the ORNL report for the RFT and the MSLB are the principal focus of this report since they raise the question of whether or not there is an immediate safety concern at Oconee or other plants. According to the ORNL report, the MSLB is the lowest probability event which has been analyzed. The MSLB is stated (small table in Section 3.1) to have an occurrence frequency of $5 \times 10^{-6}$ per reactor year. Not mentioned in the report, but apparently included in the quoted occurrence frequency (in order to produce overcooling conditions sufficiently severe to potentially fail the pressure vessel), is the probability that the operator fails to isolate feedwater to the steam generator with the broken line. That human error probability (HEP) has been multiplied by the MSLB occurrence frequency to obtain the estimated frequency of an overcooling event that would challenge the pressure vessel at Oconee-1, (i.e., the estimated frequency of the MSLB is not stated in the ORNL report, but apparently ORNL assumed the value quoted in WASH-1400 and then used a paritcular HEP to obtain the quoted value of $5 \times 10^{-6}$ per reactor year shown in the ORNL report for the overcooling event). At Babcock and Wilcox (B&W) plants other than Oconee Units 1, 2, or 3, an automatic feedwater isolation system and a steam line isolation system are installed. Proper operation of those systems following a MSLB would prevent an overcooling event severe enough to challenge the pressure vessel. Therefore, at other B&W plants, estimated frequency of this event is approximately $5 \times 10^{-6}$ per reactor year times the probability that automatic feedwater isolation fails, times the estimated frequency that the steam line isolation system fails. (Details of how the exact systems vary from plant-to-plant are not completely described in this brief summary.) This combined estimated frequency for other B&W plants would be below the value stated in the ORNL report of $5 \times 10^{-6}$ per reactor year.

In the small table in Section 3.1 of the ORNL report, the RFT is stated to have an occurrence frequency of 1.0 per reactor year.

However, additional failures would be necessary to cause a severe overcooling event as a result of the RFT. Therefore, the statement immediately below the reported occurrence frequency of 1.0 in the table must be considered as a vital qualification of that frequency, i.e., "...for Oconee-1 it appears that multiple independent failures are required..." The occurrence frequency for a mild transient initiated by the feedwater system is indeed close to 1.0 per reactor year since such transients are frequent, but such transients are of no consequence to plant safety unless there are subsequent failures. The probabilities of all the other failures must be combined in order to arrive at the actual estimated frequency of a severe overcooling event. That estimated occurrence frequency is believed to be low, as discussed below.

The estimated occurrence frequency of the particular, detailed RFT scenario presented in the report is very low since the total amount of feedwater assumed to be pumped into the steam generators is considerably greater than the maximum condensate physically available in the system at a location where it can be a source of feedwater. Assuming the amount actually available instead of the ficticiously larger amount would decrease the cooling and make the actual transient less severe. In addition, feedwater flow rates would probably be reduced below those assumed in the report, even while water is still available in the system to be pumped into the steam generators. This would normally result from loss of the steam supply to drive the turbine driven pumps as a consequence of flooding of the steam generators which are the source of that steam supply. That is, gross overfeeding of the steam generators might be self-limiting under such extreme conditions. This was not taken into account in the subject report and may be applicable to the MSLB event as well as to the RFT event.

Therefore, the NRC staff would expect that an actual RFT would be less severe than the one calculated in this report, and it would still require several failures, including: the feedwater controller or integrated control system (ICS) must fail; the BTU limiter must fail; and the operator must fail to correctly diagnose the problem and take corrective action. (These items are discussed in further detail in the body of this report.)

The above discussions of overcooling event analyses would not be complete without mention of the computer codes used. These fall into two categories, fracture mechanics codes and transient codes. The transient codes are used to calculate the pressure and temperature versus time that is input into the fracture mechanics codes; that is, they do the systems calculations that predict what pressures and temperatures will result, given a particular hypothetical event. The fracture mechanics codes assume the particular pressure and temperature versus time history calculated by the transient codes (i.e., that a particular event has occurred). These codes are then used to calculate the probability that the pressure vessel will fail if it has a certain size and shape crack present at a critical location on the inner surface.

The fracture mechanics code used in the ORNL report, together with the input data (i.e., materials properties, including fracture toughness and variations of materials properties with temperature and exposure to neutron radiation) should yield somewhat conservative results. That is, if they differ from reality, it is believed that failures would be predicted when they would not in fact occur.

Detailed evaluation of the mechanics of materials aspects of the ORNL report is difficult because of insufficient information for many of the values used. For example, Table 7.1 (page 7-1) lists parameters which must be known in order to set up OCA-1 for a thermal shock analysis but many of them were given only by inference (e.g., alpha, E, Poisson's ratio, yield and ultimate strengths) or by reference to documents (e.g., $K_{Ic}$ and $K_{Ia}$). With respect to the heat transfer coefficient, the values given in the text (page 8-4: 1000 BTU/hr-ft$^2$-F°) and in Table 8.1 (page 8-6: either 200 or 330 BTU/hr-ft$^2$-F°) are contradictory.

The ORNL reports on the HSST thermal shock experiments (TSE) in the past have failed to follow the dictates of the ASME Section XI recommendations for analysis and toughness ($K_{Ic}$ and $K_{Ia}$) determinations. The conclusions given in the ORNL reports, that TSE calculations and observations were in agreement, relied on a somewhat circular argument of mechanical property determinations. In response to a specific NRR request, reanalysis of TSE-5a by ORNL in accordance with the ASME Code showed the original analysis to result in crack extension over-estimates (Ref. 9). The NRR interpretation of the reanalysis led to the conclusion that the Code method would have added about 45°F of conservatism to the prediction.

Section 7 of the ORNL report (and, to a lesser extent, other sections) sets forth arguments of conservatism and non-conservatism in a mixed, non-rigorous way. While it is true that several uncertainties exist, most of them are recognized and accounted for by selecting individual values conservatively. Because the draft ORNL report does not present the results in terms of a sensitivity analysis, the overall degree of conservatism in the fracture-mechanics code results cannot be evaluated quantitatively.

With respect to the systems codes used to calculate the primary system coolant behavior, the IRT code used in many of the calculations is not believed to be appropriate for such calculations for several reasons. It does not contain a realistic model for flows in the primary system once significant voiding has occurred. Instead, it uses flows that are input as a table and therefore are invariable. When significant voiding occurs, as it does in the RFT event presented, the code continues to assume primary system coolant flow to the steam generators, where it is cooled and then returned to the primary system thereby making the pressure vessel overcooling event worse. In actuality, it is predicted that for certain cases the voids would collect in the system high points. If sufficient voids collect at the top of the hot leg inverted U-bend, a natural circulation flow interruption (or "vapor lock," to use a common analogy) would occur and overcooling would be greatly decreased. In this way, the code would tend to predict events more severe than the actual event. The code also has a different cause for inaccuracy in an unknown direction. That is, the code fails to conserve mass and energy throughout the calculation, by observed amounts as much as 25%. The energy or mass flowing out of one volume does not equal the total amount of energy or mass received at all other volumes as a result of the flow, as it must in reality. This discrepancy can result in errors that will vary in magnitude and direction (i.e., errors due to this latter inaccuracy can be conservative or non-conservative by varying amounts).

In conclusion, the ORNL report states that overcooling events similar to the most severe event that has occurred (Rancho Seco) will not pose a threat to the Oconee-1 vessel for several more years. The report then goes on to demonstrate that it is possible to postulate more severe events (with correspondingly lower occurrence probabilities) for which vessel failure is predicted.

The staff's judgment is that the occurrence frequency of a severe overcooling transient that would threaten the integrity of the Oconee-1 vessel is sufficiently low that time is available for the staff to carefully evaluate the condition of the vessel and propose solutions to the problem before further regulatory action is needed.

## III. Discussion

The sections below provide a more comprehensive summary of the preliminary review of the ORNL report by the NRC staff.

### A. The Overall Report from a Systems Viewpoint

#### (1) Runaway Feedwater Transients (RFT)

Of the general classes of transients identified, the most complex from a system and control viewpoint is the Runaway Feedwater Transient (RFT). The Feedwater Control subsystem of the Integrated Control System (ICS) is designed to maintain a total feedwater flow equal to the feedwater flow demand. The flow in the feedwater system is controlled by the ICS in the automatic mode by using input signals and monitoring process parameters, or it can be controlled by the operator through the ICS at the Loop Feedwater Demand level or at the Feedwater Valve Position or Feedwater Pump Speed level. The operator can intervene at any time, therefore he can be an initiator and/or a terminator of an overcooling transient.

The cause of the RFT may be internal or external to the ICS (component failure or operator error). The severity of the overcooling transient for Oconee-1 can be reduced or terminated in one of three ways: (1) The operator can take control of the FW pumps or valves, if his indication (i.e., instrumentation) is operating and he diagnoses the problem correctly, in which case the event can be terminated quickly. However, if the event is not diagnosed correctly, he may make the overcooling more severe. (2) The BTU limiter can limit FW demand since it continuously calculates the BTUs or energy contained in the steam generator. (3) The hi-level limit is a fixed setpoint which limits the liquid level in the steam generators. The hi-level trip (present only on Oconee Units 1, 2, and 3 but not on other B&W plants) can trip FW pumps at its independent fixed setpoint. All of these potential mitigation actions were implicitly assumed to fail in the runaway feedwater transient in the ORNL report.

(2) Main Steam Line Break (MSLB)

The Oconee Units 1, 2, and 3, and Rancho Seco do not have main steam isolation valves (MSIVs), whereas all other B&W units do have MSIVs. The three units at Oconee do not have automatic FW isolation, whereas all other B&W units do have FW isolation. All B&W units except the Oconee units have some type of main steam line break logic that will isolate FW to the faulted steam generator (at some plants both steam generators) and at some plants the logic will also shut the MSIVs. At most units the cooldown transient would be terminated by the steam line break logic (MSIV closure and/or feedwater isolation will terminate the cooldown, with a delay for SG dryout in the case of only feedwater isolation). It is, however, necessary to take credit for the operator terminating or throttling the HPI pumps at a later time in the event.

At Oconee-1 with no MSIVs or automatic FW isolation, operator action must terminate the MSLB overcooling transient by closing the FW valves and allowing the steam generator to steam dry. No credit was assumed in the ORNL report for operator action to limit feedwater flow for the MSLB accident.

(3) Main Feedwater System as it Would Operate for a RFT and MSLB

In the calculations of the ORNL report, full main feedwater flow is assumed throughout these events. With one steam generator flooded and the other steam generator isolated, as in the MSLB, the operator will probably not be able to maintain the turbine driven MFW pumps in a running condition because the primary steam source has been lost. The condens‚te pumps and condensate booster pumps do not have enough head to maintain full flow for these events. Multiple failures must occur in the ICS to prevent automatic runback and trip of the MFW train. Without a sufficient supply of feedwater (condensate in the hotwell) the MFW pumps will eventually lose suction. As pointed out in the Summary and Conclusion section above, the ORNL report does assume more than the actually available amount of condensate for the RFT event. All of these reasons why MFW may be lost or reduced were ignored in the ORNL report analyses, thereby making the overcooling more severe for the RFT and MSLB accidents in the report.

In addition, the feedwater temperature was assumed to ramp down to the hot well temperature within one minute after MFW pump trip, a conservative assumption. The likelihood of such behavior is extremely small since multiple failures of various systems would have to occur.

The report acknowledges that plant design modifications have already been made which will reduce the likelihood of excessive feedwater transients at Oconee-1. No attempt was made to determine the effect of these modifications on the plant's susceptibility to such transients, i.e., no credit was given for the decreased expected frequency of these transients resulting from the modifications.

B. <u>The Overall Report from a Probability and Risk Assessment (PRA) Viewpoint (Probability of Transient and Accident Sequences)</u>

(1) Summary

We have concluded that the occurrence frequencies estimated in the ORNL report for the types of initiating events analyzed are reasonable and in fact are conservative for the MSLB and

RFT when compared to estimates in use by the NRC staff. Comparison of the ORNL report and NRC estimates is given in the following table. The ORNL report does not appear to distinguish (as done below) between the probability of the initiating event and the resulting overcooling event probability.

Estimated Frequency Per Reactor Year

| | ORNL | | NRC | |
|---|---|---|---|---|
| | Initiating Event | Pressurized Overcooling Event-Resulting -from Initiating Event* | Initiating Event | Pressurized Overcooling Event Resulting from Initiating Event* |
| RFT | 1 | Not Stated | $3\times10^{-1}$ (B&W)<br>$6\times10^{-2}$ (CE&W) | $\sim10^{-4}$ (B&W)<br>$<2\times10^{-5}$ (CE&W) |
| Large MSLB | Not Stated | $5\times10^{-6}$ | $1\times10^{-4}$ | $3\times10^{-6}$ |
| Small LOCA | $3\times10^{-4}$ | Not Stated | $3\times10^{-4}$ | $1\times10^{-5}$ |
| Large LOCA | $1\times10^{-4}$ | Not Stated | $1\times10^{-4}$ | Not Stated |
| Rancho Seco | Not Stated | Not Stated | $3\times10^{-1}$ (B&W)<br>$6\times10^{-2}$ (CE&W) | $\sim10^{-3}$ (B&W)<br>$\sim10^{-4}$ (CE&W) |

*Equal to frequency of initiating event times probability of additional failures and/or error probabilities as discussed in text.

Overcooling transients at pressure in PWRs result from small break LOCAs, main steam line breaks, or feedwater transients, only if additional failures, either hardware- or human-related, occur subsequent to the initiating event. From a PRA viewpoint, we believe that a more realistic way to analyze an overcooling transient at pressure is to consider it as a sequence of events. Using event tree methodology, the overcooling transient sequence is represented by a set of event trees. Each event tree in the set has event headings, corresponding to a different initiating event, a specific assumption made in the analysis about feedwater flow rate and/or temperature, or a postulated failure (hardware- or human-related). Rigorous determination of the estimated frequency of occurrence for each event sequence thus generated would involve assigning an estimated occurrence frequency to each event and combining them to obtain the estimated event sequence frequency. Such an effort was beyond the limited scope of this review. However, within the past year, the NRC staff has made simplified analyses to obtain estimates of the frequency of overcooling transients which are summarized in the above table. To date, we are not aware of any subsequent analysis, including the subject ORNL report, that would cause us to alter those estimates.

The integrated NRR/RES task action plan being prepared for the technical resolution of the pressurized thermal shock issue includes a rigorous PRA analysis of the overcooling transient event sequences such as that discussed above.

(2) Discussion

Specific comments regarding each of the classes of initiating events are as follows:

(a) Rancho Seco Event: The most serious pressurized overcooling event was that at Rancho Seco on March 20, 1978, in which the coolant temperature dropped from 550°F to 280°F in about 1 hour while the system pressure first dropped, then returned to near its original value. Based on this experience, an occurrence frequency of $3 \times 10^{-2}$ per reactor year was estimated for a B&W plant to experience an overcooling transient as severe or more severe than the Rancho Seco event (as described in M. A. Taylor's memorandum of October 29, 1980, Ref. 3).

Since this occurrence and the occurrence of two less severe events, operators have received special training in transient response. Babcock & Wilcox plants have added a back-up power supply to the non-nuclear instrumentation bus, whose failure initiated the three transients above. The NRR staff examined the impact of the improved power supply and operator training and suggested that these improvements might have reduced the frequency to $10^{-3}$ per reactor year for an overcooling transient as severe as the Rancho Seco event for B&W plants. For more severe events, such as the RFT, that might challenge the Oconee-1 vessel if they were to occur today, the staff estimates that their frequency is $10^{-4}$ per reactor year for B&W plants.

The operating experience of CE and Westinghouse plants has also been examined. There have been no events like the Rancho Seco transient, but there have been some precursors. These are events which typically led to secondary steam dump valves or steam bypass valves sticking open, but which did not result in steam flows large enough to produce very severe overcooling transients. The most severe of these transients occurred at Arkansas Nuclear One-2 (a CE plant) on December 27, 1978, where a main steam relief valve lifted and failed to reset, thereby causing the reactor coolant temperature to drop by 107°F in 52 minutes. This lack of severe overcooling events at CE and W plants plus the greater thermal inertia of most W and CE plants, leads the staff to estimate an RFT occurrence frequency of a factor of 5 lower than for B&W plants, $2 \times 10^{-5}$ per reactor year. Also, the staff estimates the frequency of a large steam line break or its equivalent to be no greater than about $10^{-4}$ per reactor year, and for a pressurized overcooling event resulting from a MSLB severe enough to challenge the Oconee-1 vessel if it were to occur today, the estimate is a factor of 30 lower, $3 \times 10^{-6}$ per reactor year. These estimated frequencies are summarized in the above table. There may be a factor of 10 uncertainty associated with these estimates.

b. Small Break LOCA: The ORNL report does not provide complete calculations for the small break LOCA. However, in a simplified analysis of an overcooling event initiated by a small break LOCA, (i.e., between 2" and 6" equivalent diameter), the NRC staff (Ref. 5) obtained an estimated frequency of occurrence of $1 \times 10^{-5}$ per reactor year. This result was based on an assumed occurrence of $3 \times 10^{-4}$ per reactor year for the LOCA event and an operator human error probability of $3 \times 10^{-2}$ (operator failure to throttle or terminate safety injection pumps).

c. Main Steam Line Break: The table on page 3-4 of the ORNL report gives an estimated frequency of occurrence for an overcooling event resulting from a MSLB of $5 \times 10^{-6}$ per reactor year.

Reference 5 contains the results of a simplified analysis by the NRC staff of the probability of occurrence of an overcooling transient caused by a MSLB and subsequent operator error. These results are summarized as follows.

For the case of a large MSLB-initiated overcooling transient, the estimated frequency of occurrence was $3 \times 10^{-6}$ per reactor year. This result was based on an assumed frequency of occurrence of $1 \times 10^{-4}$ per reactor year for a large MSLB and an operator HEP of $3 \times 10^{-2}$ per demand (failure to terminate feedwater flow to the steam generator (SG) with the broken line and/or failure to close the main steam isolation valves to that SG).

d. Runaway Feedwater Transient (RFT): The RFT analyzed in the ORNL report assumes multiple failures subsequent to an initiating event. A better description might be given by the term "overfeed transient." Such transients usually arise from other transients which initially empty the steam generator(s), such as, in this case, a stuck-open bypass valve. Following this, a loss of automatic feedwater control or a manual error coupled with the failure of the operator to diagnose the situation and take appropriate corrective action would result in excessive feedwater being supplied to the steam generator. The NRC staff has performed a review of Licensee Event Reports (LERs) regarding overcooling events. Based on this review, a frequency of occurrence of overfeed transients of $3 \times 10^{-1}$ per reactor year was estimated for B&W plants. The corresponding estimate for Westinghouse and CE plants was $5 \times 10^{-2}$ per reactor year. A realistic estimate of the frequency of occurrence of an RFT must consider the frequency of occurrence of the initiating event and further independent failures (e.g., failure of the steam generator high level MFW pump trip) and/or continued inappropriate operator actions which exacerbate the transient. The inclusion of multiple failures, both human and hardware-related, requires analysis of an entire spectrum of RFTs. The NRC staff's action plan for resolution of the pressurized thermal shock issue includes such a complex analysis. The occurrence frequencies are believed to be low, but quantitative results are not now available. However, a preliminary estimate is given in the above table.

C. The Overall Report from a Systems Code Viewpoint

The NRC staff considers the use of the IRT computer program to evaluate the response of the primary system to severe overcooling transients to be inappropriate, since this class of events is well outside the range of the program's capabilities. IRT is capable of handling mild or intermediate transients which do not result in void formation in the primary system. IRT does not adequately conserve mass and energy.

The following critical items demonstrate the shortcomings of IRT for use in analyses of severe overcooling transients:

(1) Flow Distribution: IRT does not solve the momentum equation. The input data specifies the primary flow. The heat removal rate is dependent on the flow. For the cases presented, a natural circulation flow taken from the Oconee FSAR is input.

(2) Voids in the Primary System: In the IRT calculations, void formation is allowed only in the reactor vessel upper head region. The effect of voiding is not properly treated in IRT, which is a homogeneous equilibrium calculation. For certain of the cases presented in the ORNL report, the upper head region is voided at about 100 seconds. After this time, additional voids are incorrectly

assumed to be homogeneously mixed throughout the primary system. The assumed primary system flows in cases involving assumption of single-phase natural circulation flow are therefore incorrect. In fact, it is expected that the voids will collect in the pipes leading to the steam generators and interrupt the circulation, de-coupling the secondary system from the primary system and removing the heat sink. The loss of the heat sink will stop the cooldown.

(3) Energy Balance: For the "runaway feedwater" transient at 1000 seconds, the discrepancy in the energy balance amounts to 25% (Ref. 7). Brookhaven National Laboratory (BNL) estimates that this corresponds to a temperature error of approximately 30°F. It is not known whether the error is conservative or non-conservative. For the MSLB overcooling transient, the discrepancy is negligible.

The NRC staff asked Los Alamos National Laboratory (LANL) to perform TRAC code calculations of MSLB and "runaway feedwater" overcooling transients for conditions similar to those performed by BNL with IRT. The results from the TRAC calculations for the MSLB show that the temperatures calculated by the two codes are in agreement with one another (Ref. 8). The pressures differ, however, because of the difference in modeling of the pressurizer. A non-equilibrium model is used in IRT while TRAC uses an equilibrium model. Actual pressures would probably lie between results from the two codes.

TRAC and IRT differ greatly in their modeling capabilities of various phenomena. For example, two phase flow in the primary is modeled in TRAC and not in IRT (except in the upper head and pressurizer). An agreement between the results calculated by the two codes could be expected to occur only if single liquid phase exists in the primary.

In summary, IRT is not an appropriate program to use for severe overcooling transients. The treatment of momentum (flow) and void formation are key elements in the transient behavior. The results presented after 100 seconds are highly suspect. The staff believes the overcooling rates calculated are conservative. However, it is not possible to quantify the amount of conservatism.

D. The Overall Report from a Fracture Mechanics Viewpoint

(1) Codes

The use of the OCA-1 computer code to calculate the stress intensity (using assumed flaw morphology) is acceptable. Nevertheless, both the ORNL and the NRC staff agree that it is important to modify the code at the earliest possible date to include the temperature-dependence of material parameters (such as the elastic modulus, coefficients of thermal expansivity and conductivity), rather than using average values. Also the possibility of crack arrest in materials of high toughness (relatively high temperatures, upper-shelf energy levels) has not been addressed. Since advanced elastic-plastic fracture mechanics concepts are required in treating this matter, the lack of a solution in the ORNL report is not surprising. However, accurate calculation of the crack arrest behavior will hinge on the treatment of the high temperature, high toughness aspects.

In general, the fracture mechanics calculations in the ORNL report were performed in a manner with which the NRC staff agrees, but refinements to the OCA-1 computer code have been suggested for some time and improvements could be made today. The OCA-1 code can provide results which can be judged accurate; the results will be conservative when conservative inputs (lower bound $K_{Ic}$, $RT_{NDT}$ per R.G. 1.99 etc.) are used as was done by ORNL.

(2) Warm Prestressing

Warm prestressing (W$^2$S) is the term applied to a phenomenon which can limit the extent of total crack advance during some overcooling transients. WPS has been demonstrated in the laboratory with small specimens and in a large thick-walled cylinder during an unpressurized thermal shock experiment.

In general, the NRC staff believes that considerable caution must be used if any credit is taken for the effects of warm prestress in analyses of the pressurized thermal shock problem. The draft ORNL report does show results both with and without WPS (Table 8.6) although the comments in Table 8.7 indicate that WPS is effective only for a large-break LOCA. There are so many detailed variations in the postulated accident scenarios, involving turning pumps on or off and tapping several water sources, that the time variations in $K_I$ are quite uncertain. Only with a rather smooth change in $K_I$ relative to the toughness, $K_{Ic}$, (which also varies with time) can the benefits of WPS be assumed with confidence.

E. <u>Fluence Uncertainty</u>

The ORNL report states that the uncertainty in fluence estimates can be as great as ±50%. However, contrary to the ORNL estimate of ±50%, the staff believes that current fluence estimates can be made to within ±20% provided that one uses: (a) a well calibrated and benchmarked transport code and (b) measured values of the neutron flux and its distribution.

The staff has a threefold program for code calibration and benchmarking using: (a) the results of the PCA experiment, (b) the surveillance capsule results from Maine Yankee and Fort Calhoun, and (c) the surveillance results from ANO-1. Consequently, the staff expects future fluence calculations to be used in longer term resolution of this issue, to be within ±20% instead of the ORNL report's stated uncertainty of ±50%.

References

1. Board Notification - Thermal Shock to PWR Reactor Pressure Vessels (BN-81-06), May 8, 1981.

2. Thermal Shock to PWR Reactor, April 28, 1981, Memorandum for Harold Denton from D. Eisenhut.

3. Insights on Overcooling Transients in Plants with the B&W NSSS, October 29, 1980, Memorandum for S. Fabic from M. Taylor.

4. Pressurized Thermal Shock, Commission Paper, SECY-81-286, May 4, 1981.

5. Frequency of Excessive Cooldown Events Challenging Vessel Integrity, Ashok C. Thadani to Gus Lainas, Memorandum dated April 21, 1981.

6. Summary of Meeting with the Babcock & Wilcox, Westinghouse, and Combustion Engineering Owners Groups on July 28, 29, and 30, 1981, respectively, concerning pressurized thermal shock to reactor pressure vessels (RPV), Docket No. (All Operating PWRs), August 14, 1981.

7. October 13, 1981 letter from M. Levine to S. Fabic, "Mass and Energy Non-Conservatism in Overcooling Calcuations with IRT."

8. Personal communication to N. Zuber, RES/NRC, from LANL, about October 15, 1981.

9. January 6, 1981 letter from G. D. Whitman to Warren S. Hazelton.

Enclosure 2

DUKE POWER COMPANY

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A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

October 20, 1981

Mr. Robert M. Bernero, Director
Division of Risk Analysis
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: ORNL Evaluation of the Threat to PWR Vessel Integrity Posed by Pressurized Thermal Shock Pressure; Draft Interim Report

Dear Mr. Bernero:

Duke Power Company appreciates the opportunity to provide comments on the subject document. As you are aware, Duke has provided certain specific technical information regarding the Oconee Unit 1 reactor vessel in an effort to assist the NRC in the completion of this evaluation. Duke engineers have reviewed the subject document and consider that the evaluation contains significant deficiencies in the area of thermal-hydraulics conditions and represents unrealistic transient conditions. The application of these transients to the Oconee 1 specific material properties results in misleading and meaningless calculated vessel lifetime. Our more salient concerns are in the following paragraphs with additional details provided in the attached.

The evaluation of the reactor vessel thermal shock issue is extremely complex and requires a thorough understanding of several highly technical disciplines. Among the technical areas involved are instrumentation and controls, systems analysis, reactor vessel materials, non-destructive examination techniques, linear elastic fracture mechanics, and probabilistic risk assessment. In order to do a meaningful evaluation, these technical areas need to interact in a coordinated manner; the results of one area cannot be input into subsequent analyses without a thorough understanding of the basis of the input. This document does not indicate that any coordinated effort was attempted by the various organizations involved to assure that the results provided were realistic. In fact, the document tends to imply that the individual tasks were performed independently of each other with the end result being a totally disjointed document that is not suitable for understanding and communicating the real perspective of the issue.

One of the principal mechanisms contributing to the occurrence of pressurized reactor vessel fracture is the creation of certain unique temperature-pressure time histories at the reactor vessel, the calculation of which would require insights into plant design features, system failures and effects, plant performance constraints, and transient behavior. The fracture mechanics analyses embodied in this document are, in most part, based on arbitrary, artificial

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thermal-hydraulic accident conditions and not germane to the real plant situation, especially for the Oconee reactors. The major deficiencies in the thermal-hydraulic analyses are identified in the attachment to this letter. Portions of the ORNL report have also very appropriately discussed the limitations and deficiencies in the thermal-hydraulic analyses. Yet fracture mechanics calculations were done for these extraneous and irrelevant accident conditions.

The subject document is inconsistent within itself, which can cause significant interpretation dilemmas. The report was originally intended to be an evaluation of the B&W NSSS design and its susceptibility to pressurized thermal shock. However, the document contains statements which make it unclear as to whether or not the intended purpose was achieved as noted by the following. In Chapter 1.0, it is stated that although Oconee 1 was selected for the initial study,

> "...thermal-hydraulic behavior needs to be further evaluated as recommended later in this report and because there are special control systems provisions in Oconee-1 limiting transients, more analysis needs to be done before their results are applied to Oconee-1 or generalized to other plants."

This is further elaborated upon by the following from Chapter 5.0:

> "All the current simulations possess limitations which give concern for the realism of the thermal-hydraulic predictions. These limitations are, in part, inherent in the codes and also result from modeling deficiencies and questionable input assumptions..."

And yet the following statements occur in Chapter 8.0 without qualification:

> "A summary of results for the five overcooling accidents analyzed is presented in Tables 8.6 and 8.7. Table 8.7 indicates the total number of EFPYs that a B&W-type reactor can operate before the overcooling transients considered would likely result in vessel failure."

and also,

> "...the inclusion of cladding in the analysis will also result in smaller threshold fluences. Thus, in this respect the results in Table 8.6 and 8.7 are somewhat optimistic."

We consider these latter two statements as misleading and inappropriate considering the significant limitations of the study.

An additional concern is that the subject document does not sufficiently address significant programs currently in progress that address the areas of vessel material properties that are supported not only by Duke Power, but also by other utilities that own plants with the B&W NSSS design. This is particularly surprising because by letter dated May 12, 1981, J. Mattimoe, SMUD, on behalf of the B&W Owners Group, submitted a letter report to the Staff outlining such programs that had been completed and those still underway. By failing to recognize the other ongoing studies on this issue, the report implies that it is "the best available information." This is incorrect. It should be noted that certain branches within the NRC Staff are aware of these programs.

The evaluation of the reactor vessel fluence aspect, the interpretation of the Oconee reactor vessel material parameters, and the fracture mechanics calculations contained in this report have also several limitations. It is apparent that the chapter on fracture mechanics calculations contains several pessimistic presumptions and opinions based on unsubstantiated data and limited information. A technical report of this nature should be based on an objective analysis.

Further, the document fails to address two important items which are associated with this issue. One is the enhanced inservice examination of the reactor vessel beltline region welds in order to achieve a higher confidence level in selection of initial flaw size. As the NRC Staff is aware, such an enhanced examination was performed on the Oconee 1 reactor vessel during the current outage, using an ultrasonic technique with a stand-off distance that allows detection of near-surface flaws. Not only were all results within ASME code allowable, but also they were smaller than those sizes critical to the thermal shock issue. All indications were considered to be pre-service induced rather than service induced. The second item is thermal annealling, which is briefly mentioned, and then only in a positive sense. While the technique used in controlled conditions may seem promising, extensive work and effort will be required to perfect a technique suitable for use on an irradiated PWR reactor vessel. It is misleading to state that such a technique is currently practical, particularly when solely based on a personal communication and preliminary laboratory results.

As in the case of many other severe accidents, reactor vessel thermal shock cannot be envisioned to be forgiving to all bounding and overly conservative assumptions. In order to obtain meaningful conclusions of the severity of the problem, it is necessary to analyze systematically accident conditions by considering relevant initiating events, mechanistic system failures, and credible operator actions and by utilizing phenomenological models and methods that take into account realistic system boundary conditions and plant performance constraints. Duke has recognized that the reactor vessel thermal shock issue is a very important issue that requires careful study and timely resolution, and the way to approach the issue is by means of a cogent and systematic analysis of relevant accidents and by consideration of plant specific features both in regard to system capabilities and vessel parameters. Duke has been fervently working on such an effort, and it is our hope that when this work is completed, the necessary perspective on this matter will be obtained.

In summary, the report in its present form is not suitable for understanding and communicating the real perspective of the issue. In fact, it could unduly distract attention from the orderly efforts now being pursued on the resolution of the issue. Accordingly, we ask that the report be modified significantly taking into account our comments or be withdrawn from general release.

Very truly yours,

A C Thies

A. C. Thies

RLG/php
Attachment

cc: Mr. R. C. Kryter
Instrumentation and Controls Division
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DUKE POWER COMPANY

Detailed Comments on
ORNL Draft Interim Report
Evaluation of the Threat to PWR Vessel Integrity
Posed by Pressurized Thermal
Shock Events

## Chapter 1.0

page 1-2, 3rd paragraph:

We agree with the statement about the need to perform "realistic systems analyses to determine appropriate input temperature and pressure transients for the vessel integrity studies, and [to evaluate accurately] the mechanical integrity of the pressure vessel" through plant specific studies. However, the analyses conducted thus far fall short of this goal, as recognized on page 1-3:

> "...because thermal-hydraulic behavior needs to be further evaluated as recommended later in this report and because there are special control system provisions in Oconee 1 limiting transients, more analysis needs to be done before their results are applied to Oconee 1 or generalized to other plants."

## Chapter 2.0

A clear and consistent definition of the "runaway feedwater transient" is necessary. The thermal-hydraulic analyses utilized in this report consider this transient to consist of an unmitigated main feedwater overfeed transient event with a concurrent failure of the turbine bypass valve system following a reactor trip transient. However, the probability discussion of Section 3.1 apparently visualizes this accident as a more general secondary system upset condition which includes steam generator overfeed transients, steam generator pressure control malfunctions, and events involving failures in feedwater flow control and SG pressure control functions.

## Chapter 3.0

page 3-1:

In order to obtain the real perspective of the safety significance of this problem, one needs to consider the probability of occurrence of a break in the reactor vessel at the correct location and of sufficient size to compromise adequate core cooling capability as a result of crack initiation and propagation. This probability is composed of several (possibly independent) probabilities, including (1) the probability that a break large enough would occur given that the fracture mechanics calculations predict a through-wall crack propagation, (2) the probability that a through-wall crack propagation would occur given the specific pressure-temperature condition (this probability is dependent on the probability that flaws of certain unique size and orientation capable of through-wall propagation exist at the location of minimum material strength), and (3) the probability that the potential transient events produce the pressure-temperature conditions necessary for unarrested crack propagation.

page 3-1, Table and 3rd paragraph:

No basis is provided for the assigned probability of a runaway feedwater transient (RFT). The value provided is arbitrary and is not based on any review of operating experience or quantitative assessment of probability of RFT that causes severe overcooling conditions. The RFT characterized by a frequency of occurrence of 1/Ry represents a general secondary system upset condition of an overcooling nature and not the accident treated in the subsequent sections of the report.

page 3-3, 1st paragraph:

The EFPY results provided in this paragraph are not valid due to the inherent errors and limitations of the thermal-hydraulic conditions utilized. Furthermore as discussed in detail later, the assumed fluence rate per EFPY is inaccurate.

page 3-3, 2nd paragraph:

The basis of this statement is not apparent. Figure 5-4 shows predicted temperature response for all transients including RFT and MSLB (IRT). Within 600 secs for RFT and 250 secs for the MSLB, primary coolant temperature is predicted to be below 200°F. This figure would tend to indicate that the predicted temperature is well below 212°F during most of the transient rather than well above 212°F at the time of predicted failure.

## Chapter 4.0

page 4-1, 2nd paragraph:

The last sentence is incorrect. The neutron power signal obtained from the RPS can modify main feedwater demand if its mismatch with the ICS reactor demand level exceeds a set tolerance only if the conditions in the steam generator permit, i.e., BTU limits, high and low S.G. level limits override. Loop A and B steam generator feedwater demands are reduced to zero in 15-20 seconds following reactor trip due to the combined actions of cross limits and BTU limits. Tripping of RC pumps due to HPI actuation also requires that the operator verify the reactor has tripped. When the reactor trips, the ICS controls feedwater flow as described above.

page 4-1, 3rd paragraph:

The section is entitled "Reactor Protection System." The integrated control system (ICS) discussed in the paragraph is not part of the RPS and should be separated out.

A reactor trip will not only occur upon turbine trip, but also will occur on loss of main feedwater.

The RPS low pressure trip is 1800 psi.

page 4-2, 3rd paragraph:

The Low Pressure Injection System is incorrectly described. Only two LPI pumps are started automatically. The third pump can be manually started and aligned to either A or B train.

Although the core flood tanks (accumulators) are mentioned in Sections 4.4.2, 4.4.3, there is no description in the system description paragraph.

page 4-2, 6th paragraph:

The LPI System is actuated when the primary system falls below 500 psi. Substantial flow, however, by this system, could occur only when the system pressure falls below 200 psi.

page 4-2, 7th paragraph:

While the discussion of the main feedwater control is fairly accurate, the discussion fails to include any mention of turbine bypass valve control and only briefly discusses features of the ICS that tend to limit the potential for an overcooling event.

Further, there is no mention of the Emergency Feedwater System and its controls and instrumentation, which, in fact, are totally independent of the ICS.

page 4-5, 2nd paragraph:

The sentence is incorrect as written. The main feedwater pumps are supplied water from the condenser hot well through three condensate booster pumps and three hotwell pumps. The surge tank and condensate storage tanks provide makeup to the hotwell, not directly to the feedwater pumps.

page 4-5, 4th paragraph:

Although the maximum inventory from all sources is 295 x $10^3$ gallons, the actual usable inventory is 142,000 gallons in the hotwell. Although condensate makeup to the hotwell can be achieved from the UST and CST, the maximum condensate available for an uncontrolled main feedwater flow event is 192,000 gallons (or for 9 minutes at full flow rate).

page 4-5, 5th paragraph:

The first setpoint is incorrect. The total feedwater demand will run back at a maximum rate of 20% per minute to track generated megawatts following a reactor trip-<u>if the conditions in the steam generator will permit that demand</u>. If the steam generators cannot accept a 20% per minute runback, the BTU limits will reduce the demand to whatever value is appropriate.

The second setpoint is partially described correctly; the following should be added. The feedwater valves will transfer to emergency level control which compares the actual level in the steam generator to a 50% level setpoint. This circuitry will either open or close the startup valve as appropriate with the pumps controlling on D/P.

In addition to the listed trips, each main feedwater pump will trip on low suction pressure or on overspeed.

page 4-6:

An attempt is made to represent functionally the main feedwater portion of the ICS. This figure should be redrawn to represent more accurately the control system. As a minimum the level limiter should be moved above the controller and another controller added to control the startup valve on loss of all RC pumps.

page 4-7:

For single control failures occurring below the manual control points also, the high level trip of the main feedwater pumps will be available to mitigate the events.

No discussion of the availability of instrumentation and controls is presented. A description of the present system was provided to ORNL (copy of July 23, 1981 letter of William O. Parker, Jr. to NRC) and yet no mention is made of the multiple instrumentation available to the operator.

The first sentence on page 4-7 should be changed as follows: This review divided the <u>main feedwater portion of the ICS</u> into three general areas, as shown in Figure 4-3.

page 4-7, 2nd paragraph:

In the second sentence, manual control is required following ICS failure.

Sections 4.5.4 through 4.9 use the term excessive feedwater on numerous occasions with no attempt to define the amount of excess. Someone who does not know the system may not understand the differences and in fact could interpret excessive feedwater to mean the hypothetical runaway feedwater transient. This should be clarified in future reports.

The last sentence should be changed as follows: It should be noted that without the steam generator high level trip for the feedwater pumps, failure of <u>a</u> startup level signal to a "low" condition can result in an overfeed of <u>one</u> steam generator.

page 4-8:

In Table 4-1, it should be noted that several indicated failures cause overfeed to only one steam generator.

The Oconee 1 event sequences referred to were submitted to the Staff in July 1981 as part of the Abnormal Transient Operating Guideline Program. These are currently under review by the Staff.

page 4-10, Section 4.7:

Overcooling transients are alerted to the operator by numerous alarms and are easily recognized by decreasing temperatures and the cause identified by steam generator conditions.

The continuance of main feedwater at 100% flow rate requires multiple ICS failures and failures of other flow limiting functions or deliberate operator action to open feedwater valves to both steam generators and to disable certain trip functions. Even then, the condition can persist only for a short duration because it is self-limiting (due to high SG pressure conditions or due to rapidly diminishing inventory).

page 4-11, 2nd and 5th paragraphs:

Based on a detailed review of the IRT and TRAC calculations, we believe that to characterize them as being "approximately bounding" is overly optimistic.

<u>Chapter 5.0</u>

Of the four Oconee events, only two events can be considered as representative initiating events of the general secondary system upset condition category of events of interest in reactor vessel overcooling. These two events are the 1/4/74 switchyard isolation event of Oconee 2 and the November 10, 1979 loss of ICS power event of Oconee 3. In the Oconee 2 transient the overcooling was caused by excessive steam load combined with a high initial design prescribed steam generator level, which has subsequently been reduced. For the Oconee 3 event also, the major contributor was excessive steam load (auxiliary steam drawdown and partially open turbine bypass valve) with some minor contribution from overfeeding one steam generator. In both cases the primary system cooldown was limited to 420°F, and even if the operator had failed to take action the transient would have progressed only to a modest overcooling event and not of the severity calculated to occur in the present analyses. The third Oconee event (June 13, 1975 event in Oconee 3) involved a stuck-open PORV, and the actual thermal-hydraulic transient behavior was milder than the calculated small break LOCA transient. The fourth event involved a temporary undercooling in Oconee 1 on December 14, 1978. During this type of an event, the primary system undergoes a rapid but <u>finite</u> cooling of the primary side when normal cooling is reestablished. The primary system cooldown is limited to 520 - 540°F and as such is not different from typical reactor trip events as far as overcooling events of interest for reactor vessel integrity are concerned.

It is worthwhile to examine the operating history of the Oconee reactors with regard to the occurrence of the "RFT" event, which is characterized by the failure of the main feedwater flow control system to run back feedwater flow after a reactor trip, followed by the failure of the SG high level trip of the MFWP's and concurrent stuck-open failure of the TBV System, and not considering any operator actions. The three Oconee units combined have now accumulated 23 reactor years of operation, during which time 186 reactor trip events have occurred. Our review of these reactor trip events indicates that in all cases the feedwater was run back, either promptly or with acceptable delay, after the reactor trip and did not represent a perpetual full flow condition. Furthermore, the SG high level trip of the main feedwater pumps have been challenged nine times as a result of moderate overfeed conditions due to slow feedwater runback or during loss of ICS power events. In all cases successful trip of the system occurred as designed. With regard to the turbine bypass valve system, we have had no instances in which all the turbine bypass valves stuck open. Although we have had a few instances involving excessive steam loads and/or partial failure of the TBV System, these events produced only modest overcooling of the primary system. In all cases successful and timely operator action has been found to occur. Additionally, it should be pointed out that design changes have been made and operating procedures have been written to prevent/reduce the probability of steam generator overfeeds (RFT). The present response to all three of the overfeeds listed in Appendix B would be a trip of the main feedwater pumps which would automatically initiate auxiliary feedwater. Auxiliary Feedwater would maintain steam generator level at 25" (240" if all the RC pumps trip), thereby preventing both steam generator dryout and overfill. Operator confusion would not result on loss of ICS power since adequate backup instrumentation and controls and emergency procedures are available.

page 5-7, 1st paragraph:

Fluid mixing between the HPI and cold leg is of minimal importance during overcooling transients. The temperature on the downcomer RV is affected primarily by the temperature of the fluid at the weld location of interest and the fluid flow rate which govern the heat transfer coefficient.

page 5-7, 5th paragraph:

Flow distribution is important in determining the rate of heat removal from the RV wall and thus the temperature gradient in the wall. The assumption of an arbitrary flow affects not only the thermal-hydraulic calculation but also the heat transfer from the wall.

page 5-7:

Additional deficiencies in the thermal-hydraulic predictions beyond those identified in Section 5.3.2 are evident and are as follow:

A. It is inappropriate to use the IRT code for any external and released applications since the code is still under development. This is evident by the fact that the code does not have a momentum equation and therefore all the flow rates in the analyses are non-mechanistic. In addition, the code has not been widely used in the industry and its capabilities have not been demonstrated.

B. The justification for using IRT to simulate a B&W configuration has not been established. Once a code has been verified (this has not been completed r IRT), the nodalization of the system being modeled must be qualified by comparison with data from the system being modeled. There is no indication that this has been done using IRT on a B&W plant.

n example of this is the apparent failure to consider reactor vessel upper head circulation flow in the analyses and, also, the failure to consider the feedwater injection location and the pre-heating in the steam generator.

C. There is no indication that the analysts had the necessary intimate familiarity with the Oconee plant to set up a realistic and appropriate set of boundary conditions for a simulation. Overcooling transients are strongly affected by boundary conditions. Without a realistic set of conditions, the transient response will not represent the true response, and the results are essentially meaningless. A lot of simulation experience in terms of plant system familiarity and knowledge of code capability and limitations are essential.

Some examples of boundary condition errors are:

a. Incorrect feedwater flow
b. Incorrect turbine bypass setpoint and capacity.
c. Incorrect HPI actuation setpoint and flow versus pressure.
d. Feedwater enthalpy versus integrated flow delivered.
e. There are no secondary steam relief val es or atmospheric dumps.
f. Omission of control system responses or additional assumed failures that are not identified, e.g., high SG level trip of both main feedwater pumps.

D. It is very misleading to label a particular analysis without explicitly identifying the failure assumptions made in the analysis. As an example, the IRT analysis labeled, "Turbine Trip", is actually a turbine trip with a failure of the main feedwater to run back, with a failure of the high SG level trip of the main edwater pumps, with a failure of the turbine bypass valves on both steam generators, assuming a rapid decrease in feedwater temperature, and assuming a failure of the operator to terminate the overcooling or perform any other mitigative action. The assumptions which determine the transient response should not be lost in the generation of plots of results, and neither should the limitations of the code utilized.

<u>Chapter 6.0</u>

page 6-1, Section 6.1, last paragraph:

Although the variations in Table 6.1 do occur in source parameters, they are not necessarily uncertainties in the calculation of fluence. Many of these items are accounted for in the calculational procedure. For example, cycle and cycle-to-cycle core power distributions are averaged over the capsule irradiation period with the use of PDQ generated power distribution data at selected time intervals during fuel cycles.

The basis of these statements is experience in the analysis of 12 capsules from 8 B&W reactors.

page 6-1, 6-2, Section 6.2.1, 1st paragraph:

Although this procedure was used to calculate fluence from Oconee 1 capsules OCIF and OCIE, an improved procedure is presently being used which incorporates the capsule geometry and $P_3$ scattering cross sections directly into the r-Θ reactor model, thereby eliminating the need for corrective factors.

page 6-3, Table 6.2:

An important step was omitted, that of normalization of calculated flux to flux derived from measured dosimeter activities.

This table should read:

j. Calculate capsule flux (E>1 MEV) by multiplying the value from the r-Θ model times the $P_3/P_1$ and capsule perturbation factors and times an axial shape factor based on the axial power shape in a peripheral fuel assembly.

k. Obtain a normalization factor from the ratio of flux (E>1 MEV) derived from dosimeter reactions to calculated flux (E>1 MEV) in the capsule.

l. Perform an axial 2-D, $P_1$, r-z calculation.

m. Correct flux values from the r-Θ model with the $P_3/P_1$, capsule perturbation, axial shape, and normalization factors.

n. For weld locations, displacement factors from the r-z model and r-Θ model are applied to the vessel flux (or fluence).

page 6-5, 6-6, Section 6.2.3, last paragraph:

The spread in normalizing factors is misleading with respect to calculational uncertainties because only fission reaction data from the OCIE capsule were used to calculate fluence. Data from OCIF were discounted because of suspected errors in activity measurements. This was the first capsule analyzed at B&W and such large discrepancies have not been observed in any subsequent capsule analysis.

page 6-6, Section 6.3, first paragraph:

The uncertainty evaluation in BAW-1485 was primarily based on conservative estimates with relatively little experience. Thus values of ± 30% for predicted beltline region fluence and ± 50% for certain weld locations were reported. Since then, B&W has participated in the Blind Test, a calculational benchmark sponsored by the Light Water Reactor Pressure Vessel Dosimetry Improvement Program (NRC funded) and the OCIF fluence calculation has been checked by another phase of the LWRPVDIP. (R. L. Simons at HEDL did the analysis.) The Blind Test indicated that the B&W transport calculational procedure would produce a fast flux (E>1 MEV) that deviated ≤ 5% from a normalized capsule location to vessel surface and T/4 locations. The HEDL calculation of capsule fluence was 5% greater than the B&W calculated value. In addition, analyses of 12 capsules from 8 B&W reactors have consistently shown E/C values within ± 10% for fission reactions. Based on these developments, recent estimates of fluence uncertainties are ± 10% at the capsule, ± 15% in the vessel for time periods corresponding to capsule irradiation periods, and ± 18% for predicted fluence in the future. Comparable values for vessels in reactors without capsules are ± 18% and ± 21%. It must be emphasized that these are conservatively estimated values in the absence of a detailed uncertainty analysis.

The ± 50% value reported in BAW-1485 for weld locations was intended to indicate the added uncertainty (above ± 30%) of using axial and azimuthal displacement factors. Apparently, this was misunderstood in the ORNL analysis. A displacement factor of .89 (as is used for the critical weld location in the ORNL analysis) cannot be in error more than -12% when compared to the beltline region fluence. When statistically combined with the vessel uncertainty of ± 30%, this would result in a ± 32% uncertainty.

pages 6-6, 6-8, Section 6.4:

The implication that there has been no verification of the B&W calculational procedure for the determination of fluence is incorrect. In fact, B&W has successfully participated in the Light Water Reactor Pressure Vessel Dosimetry Improvement Program to benchmark both the calculational procedure and the dosimeter measurement technique.

page 6-7, Table 6.5:

Data in this table apparently are based on extrapolation of the fast flux averaged over cycles 1 and 2. To obtain more realistic values, the extrapolation (in time) should be based on a predictive procedure described in BAW-1485. For Oconee 1, the use of this procedure is particularly important because of a conversion to an 18-month fuel cycle in cycle 6 with a corresponding reduction in ex-core fast flux of approximately 30%.

| Inside of RV Wall | T/4 | 3T/4 | Outside of RV Wall |
|---|---|---|---|
| 2.20E+18 | 1.22E+18 | 2.86E+17 | 1.053+17 |
| 2.67E+18 | 1.48E+18 | 3.47E+17 | 1.27E+17 |
| 3.26E+18 | 1.81E+18 | 4.23E+17 | 1.55E+17 |
| 3.66E+18 | 2.03E+18 | 4.75E+17 | 1.74E+17 |

Basis is assumption that relative effect of 18-month cycle in ANO-1 will be the same in Oconee 1. Predictive data are available for Oconee 1 through cycle 7 but the calculations have not been made.

## Chapter 7.0

page 7-1, Section 7.1, Table 7.1:

Detailed descriptions of all data used and certification that such data are appropriate for those analysis have not been provided. Also, error analysis for input date has not been provided.

page 7-3, first paragraph, next to last sentence:

The basis of the statement that uncertainty is not large in the parameters included in ASME Section III is not provided.

first paragraph, last sentence:

This sentence conflicts with the previous statement. Data should be to support this position. Explanation should be given as to how such data relate to the $K_{IR}$ data which are used to evaluate vessel integrity. Also, data for the uncertainty in the determination of $RT_{NDT}$ should be provided.

second paragraph, first two sentences:

These two sentences appear to be in conflict. They should be clarified and supported with actual data to substantiate the opinion expressed in this paragraph.

second paragraph, last sentence:

The reference to support this statement should be provided.

third paragraph, first sentence:

The reference to support this statement should be provided.

third paragraph, last sentence:

This statement does not recognize the Oconee Unit 1 and the B&W Owners Group Research Program which is in progress and is generating this data.

page 7-4, second paragraph:

This paragraph appears to express an opinion which should be based on sound data. Since the statement is made that Regulatory Guide 1.99 is "not excessively conservative" for Oconee 1 weld metals, the data should be either presented or referenced so that a better definition of "excessively conservative" can be better understood.

second paragraph, last sentence:

Over 35 surveillance capsules have been removed from power reactors and the data support the conservatism of Regulatory Guide 1.99. As for irradiation programs at test reactors, most of these are completed and the data are available.

page 7-5, third paragraph:

The statement regarding reduction in upper-shelf energy of weld cladding is misleading. No mention is made of the fact that these fluences are well above that expected at EOL of any operating PWR. At the fluence levels predicted, the cladding is expected to lessen the degree of crack propagation.

page 7-5, Reference 8:

This information should be included in the analysis and should not be stated as a reference since it represents a significant reduction in EOL fluence.

## Chapter 8.0

page 8-2, first paragraph:

It is stated that if the temperature of a major portion of the coolant in the primary system is above 212°F, the opening due to crack propagation may be excessive and core cooling not maintained. It is interesting to note that RFT and MSLB transient show bulk temperature decreasing to below 200°F.

Even with the considerable errors in the transients analyses provided, it could be postulated that the copious amounts of LP injection available would be more than sufficient to maintain the core cool, in much the same way as it is predicted to occur during a postulated LBLOCA.

page 8-2, second paragraph:

It should be noted, again, that only the vessel material properties are approximately representative of Oconee 1. The accidents analyzed are not at all representative of Oconee 1 or any other plant with B&W NSSS.

fourth paragraph:

For overcooling events, little if any vent valve flow will occur because there is minimal differential pressure between the core outlet and inlet. The thermal analysis is dependent on the downcomer temperature <u>and</u> the flow conditions. It is not apparent what flow conditions were assumed in the thermal-hydraulic calculations and thus what was assumed in thermal analysis of the RV wall.

page 8-3, second paragraph:

The ORNL analysis ignores axial gradient in fluence. Axial gradient of fluence is utilized in the B&W analyses.

The assumption of a pre-existent long sharp crack is unrealistically conservative. This is particularly true for Oconee 1 which recently underwent a 100% examination of beltline region welds with no cracks indicated.

page 8-4, third - fifth paragraphs:

No basis is provided to support the assumption that the fluid-film heat-transfer coefficient is 1000 Btu/hr-ft$^2$.°F. It is stated that this corresponds to full-flow conditions, but it is not stated what flow conditions were actually assumed in the transient calculations. It is inconsistent to assume one mode of system operation during the transient calculation and a heat transfer coefficient based on a different mode of operation. This is particularly important in that severe overcooling transients may interrupt RCS flow and thus reduce heat transfer; and with RC pumps assumed running, a finite amount of heat is in fact added to the RCS.

page 8-6, Table 8.1:

The ORNL analysis should have utilized actual weld parameters inasmuch as these data were provided by Duke. The $RT_{NDT}$ is that of the base metal rather than the weld and the chemistry is that of a hypothetical weld metal.

page 8-13:

It is inappropriate to perform the fracture mechanics analyses of the IRT steam line break of RFT cases with all their known deficiencies and atypicalities. The analysis, discussion and results for these two cases should be deleted.

page 8-16, Section 8.4, last paragraph:

The fluence rate of $.046 \times 10^{19}$ n/cm$^2$/EFPY is appropriate for the critical weld location (lower long weld at inner surface of vessel) for the first 4 EFPY (through cycle 5). Thereafter, because of the conversion to the 18-month fuel cycle, a better value for >4EFPY exposure is $.033 \times 10^{19}$ n/cm$^2$/EFPY at the critical weld location.

Also, as noted previously, the ± 50% uncertainty does not apply to welds close to the beltline region. Based on the initial analysis at the time BAW-1485 was written, this value would be ± 32%; more recent estimates indicate about ± 20%. These are estimates because a detailed uncertainty analysis has not been performed.

page 8-17, Table 8.6:

The column with WPS and without WPS appears to have the line item designations reversed. WPS should provide additional time to fracture in all cases except LBLOCA. Table 8.7 appears to have this relationship correctly identified.

page 8-19, Table 8.7:

The revised values of fluence/EFPY to account for the 18-month fuel cycle will lengthen the threshold time calculation for times ≥EFPY.

| Threshold Time[a] (EFPY [b]) | Comments and Qualifications |
|---|---|
| 26 | . . . . . |
| 26 | WPS not assumed effective (44EFPY if it were). |
| | Through-wall crack predicted. |

[b] Based on fluence accumulation rate value of $0.046 \times 10^{19}$ n/cm$^2$/EFPY for the initial 4EFPY and $0.033 \times 10^{19}$ n/cm$^2$/EFPY thereafter. This value may have an uncertainty of as much as ± 32%.

## Chapter 9.0

page 9-1, -2, Mitigative Measures:

In view of the inherent weakness contained in the report, it is considered that identifying the need for any changes is premature.

As a point of clarification, increasing the BWST temperature would have essentially no effects on reducing the severity of an overcooling transient. Because a SBLOCA was not performed, it is not clear what basis there is for stating that an increased BWST temperature would reduce the degree of overcooling caused by actuation of HPI. In fact, with vent valve flow and plant specific analyses, the thermal shock concern for SBLOCA is minimized.

page 9-2, sixth paragraph:

The practicality of in-place annealling is overstated. While the principle may have been demonstrated under controlled laboratory conditions, extensive work is necessary to implement in-place annealling on an operating reactor vessel. Extensive evaluations are necessary to demonstrate the acceptability of annealling at temperatures of 750°-850°F if a plant and its support systems were designed to lesser temperatures.

page 9-2, Section 9.2.2, General Changes:

An additional change that is presently occurring in B&W plants that will significantly lower the fluence on the reactor vessel wall, is the result of going to 18-month LBP reload cycles. Once-burned fuel is loaded on the periphery of the core thus lowering the peripheral fuel assembly's power and the corresponding leakage flux or fluence to the vessel. The results of this are noted in the references of Chapter 7, Reference 8.

## Chapter 10.0

page 10-1, Section 10.1:

See previous comments on fluence analysis.

Section, Concluding Remarks:

All remarks are negative in con[illegible] and are leading to uncertainty and lack of confidence in the final results. Yet, statement is made that, "Nonetheless, for all their shortcomings, the analyses at hand are the best presently available on a nonproprietary basis, and...[merit] a great deal more study using refined techniques."

page 10-2, Section 10.2:

The implication here is that fluence calculations in general have uncertainties in the range of ± 30% - ± 50%. The uncertainty in the Oconee 1 fluence calculations is much smaller as discussed in the comments on Chapter 8.0.