UNITED STATES
NUCLEAR REGULATORY COMMISSION
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MEMORANDUM FOR: Atomic Safety and Licensing Board for TMI-1 Restart

FROM: Thomas M. Novak, Assistant Director for Operating Reactors, Division of Licensing

SUBJECT: BOARD NOTIFICATION - TMI-1 RESTART HEARING (BN-81-31)

This notification concerns the integrity of reactor pressure vessels when subjected to thermal shock and subsequent repressurization during an overcooling transient.

Enclosed is a copy of the October 9, 1981 memorandum from William J. Dircks (EDO) to the Commissioners concerning the "Status Report on Pressurized Thermal Shock." This memorandum enclosed an interim report by the Oak Ridge National Laboratory (ORNL) on a NRC research program on pressurized thermal shock which presents a preliminary assessment of the threat of pressure vessel failure.

The ORNL results of preliminary analyses predicted failure of the Oconee 1 pressure vessel for the Rancho Seco type transient of 1978, turbine trip with stuck-open bypass valve, and main steam line break. The calculated threshold times for failure were 20, 3, and 4 EFPYs respectively.

The owner of the plant, Duke Power Company, was asked to review the analysis for accuracy and a joint NRR/RES team is reviewing the report. The team plans to complete its assessment with a draft report regarding the validity of the ORNL report within the next two weeks. We will notify the Board of the results of this review.

"ORIGINAL SIGNED BY"

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Enclosure:
As Stated

cc w/enclosure: See next page

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
(Three Mile Island, Unit 1)	)	

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

October 9, 1981

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

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: STATUS REPORT ON PRESSURIZED THERMAL SHOCK

In information paper SECY-81-286 on pressurized thermal shock of pressure vessels and in the subsequent briefing of the Commission on June 11, 1981, the Commission was informed that Oak Ridge National Laboratory was preparing a status report on pressurized thermal shock. An interim report focusing on Oconee-1 has now been completed; a preprint copy is enclosed.

As noted in the enclosed report, the purposes of this ORNL work are to identify what is presently known about this problem including major areas of uncertainty and sensitivity, to identify further information needs, and to propose and evaluate possible mitigative measures. This interim report organizes what is presently known and presents a preliminary assessment, based on present analysis, of the threat of pressure vessel failure. Although the nominal calculations indicate a proximate threat, the report points out a number of flaws in the current thermal-hydraulic analysis which reflect a lack of realism. The owner of the plant, Duke Power Co., was asked to review the analyses for accuracy; the Duke representatives also challenged the validity of currently available analyses.

In order to evaluate the conclusions of the report and to assess their significance, a joint NRR/RES team has been set up to review the report as soon as it arrives in NRC. In particular, the staff will evaluate the probability of occurrence of the severe overcooling transients assumed and

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the conservatisms used in other portions of the Oak Ridge analyses. The team will be led by the NRR task manager, Roy Woods, and it plans to complete its assessment with a draft report in 2 weeks.

We will keep the Commission informed of the results of this review.

William J. Dircks
Executive Director for Operations

Enclosure: NUREG/CR-2083, Evaluation of the Threat to PWR Vessel Integrity Posed by Pressurized Thermal Shock Events

cc w/encl: PDR

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Instrumentation and Controls Division

EVALUATION OF THE THREAT TO PWR VESSEL INTEGRITY POSED BY PRESSURIZED THERMAL SHOCK EVENTS

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INTERIM REPORT

Evaluation of the Threat to PWR Vessel Integrity Posed by Pressurized Thermal Shock Events

Task Coordinator: R. C. Kryter[1]

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1.0 INTRODUCTION

Pressurized water reactors (PWRs) are susceptible to certain types of hypothetical accidents that, under some circumstances, including operation of the reactor beyond a critical time in its life, could result in failure of the pressure vessel as a result of extensive propagation of crack-like defects in the vessel wall. Accidents of particular concern are those that result in rapid cooling (thermal shock) of the inner surface of the reactor vessel (RV) wall, particularly if they also involve substantial primary-system pressure. [Such accidents have been referred to as "overcooling accidents" (excessive cooling for a particular pressure) and/or "pressurized thermal shock."]

For a particular accident and operator and system response, the tendency for preexistent vessel flaws to propagate during thermal-shock loading conditions is a function of the relative magnitudes of the stress field or stress intensity factor (K_I) and the material fracture- and arrest-toughness values (K_{IC} and K_{Ia}). These toughnesses decrease with decreasing temperature and increasing fast-neutron fluence, and K_I increases with increasing stress and is greater for a surface flaw than for a buried flaw. Thus, flaws on the inner surface of the RV wall are of greatest concern for thermal-shock loading.

The positive gradient in temperature that exists within the wall during a thermal transient and the negative gradient in fluence together result in a positive gradient in K_{Ia} that provides a mechanism for arrest of a fast-running crack. However, if the primary-system pressure is high enough, the gradient in K_I may be such that

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arrest will not take place, and the flaw will then extend through the vessel wall. Depending on the temperature and pressure of the primary system and the length and orientation of the flaw at the time of its wall penetration, the opening produced could either be negligible in size or sufficient to preclude adequate cooling of the reactor core. For instance, previous overcooling-accident calculations[1] indicate that in the event of a double-ended pipe-break loss-of-coolant accident (LOCA), which produces perhaps the most severe of all thermal shocks but also a very low vessel pressure, flaws presumably will not be driven through the wall. In another case[2] the internal pressure remains rather high, the coolant temperature remains well above saturation for atmospheric pressure, and RV failure with a sizable opening is predicted.

As already mentioned, the tendency for crack propagation increases with increasing reduction in material toughness and thus with increasing fluence. An additional factor that influences the extent of the radiation-induced reduction in toughness of present-day reactor pressure vessel materials is the presence of impurities such as copper and, to a lesser extent, phosphorous. Within limits, the higher the concentration of these two elements the greater the radiation-induced reduction in toughness for a given fluence. In the context of calculated flaw behavior under pressurized thermal-shock loading conditions, a broad range of copper concentrations exists among PWR pressure vessels currently in service. Some of the vessels in the high-copper category appear, on the bases of selected hypothetical accidents, assumed initial flaws, and presumably conservative analyses, to be susceptible to failure at early dates, while vessels with low copper category are not susceptible to failure for an extensive period.

Because of the apparent severity of overcooling accidents and the obvious complexities associated with defining accidents and their likely frequency of occurrence, performing realistic systems analyses to determine appropriate input temperature and pressure transients for the vessel integrity studies, and accurately evaluating the mechanical integrity of the pressure vessel, thorough plant-specific studies are in order.

In May 1981, the U.S. Nuclear Regulatory Commission (NRC) requested assistance[3] from the Oak Ridge National Laboratory (ORNL) in attaining such an understanding of the severity of the threat posed by pressurized thermal shock occurrences, subject to the constraint that an interim report which would consolidate, evaluate, and summarize all the pertinent data and analyses identified and collected must be produced in four months time. This short time frame precluded undertaking new studies and calculations of significant magnitude, so the evaluated results cited in this report are necessarily drawn from known previous work and literature search.

The major goals of this ORNL integrative effort were to (1) identify what is presently known about the pressurized thermal shock problem, including the major areas of uncertainty and the sensitivity of the estimated severity of threat to these uncertainties; (2) identify what is not known about the problem, including suggested means for correcting any such deficiencies; and (3) propose and evaluate possible mitigative measures. The work required to meet these goals was divided into six principal tasks:

1. Define the problem elements that dominate in establishing the overall likelihood of RV failure and develop a scheme for assessing the relative safety significance and likelihood of occurrence for the spectrum of possible initiating and subsequent events.

2. Review presently existing thermal-hydraulic analyses of various postulated overcooling scenarios and critically assess their realism and usefulness in defining a generic spectrum of overcooling events. Identify critical assumptions and input uncertainties and estimate their probable effects on the predicted temperatures and pressures.

3. Review the function of plant-specific control and safety systems, along with procedure-directed operator actions. Consider system modifications which would help to lessen the severity and frequency of overcooling transients.

4. Estimate the overall severity of threat to RV integrity imposed by pressurized thermal shock occurrences.

5. Propose potential corrective actions which might be effective in reducing the severity of threat. Discuss probable effectiveness and relative ease of implementation.

6. Provide recommendations for extending the study in an effective manner in FY 1982 to obtain a broader, more balanced understanding of the problem as it relates to the spectrum of current plant designs.

The selection of the first representative plant to be studied was somewhat arbitrary but in consideration of an extensive history of thermal-hydraulic upsets in Babcock and Wilcox (B&W) plants and the low thermal inertia provided by the B&W once-through steam generator design, a reactor built by this manufacturer seemed a reasonable choice. Since Oconee-1 has a RV with longitudinal welds having a relatively high copper content, is the lead B&W plant (commerical operation began in July 1973), and has a larger cumulative power history (~4.9 EFPY to date) than its sister units, this plant was selected (with NRC concurrence) to provide a basis, so far as practical, for our initial study. On the other hand, because thermal-hydraulic behavior needs to be further evaluated as recommended later in this report and because their 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.

REFERENCES - CHAPTER I

1. R. D. Cheverton, S. K. Iskander and S. E. Bolt, Applicability of LEFM to the Analysis of PWR Vessels Under LOCA-ECC Thermal Shock Conditions, ORNL/NUREG-40 (October 1978).

2. R. D. Cheverton and S. K. Iskander, "Thermal-Shock Investigations Heavy-Section Steel Technology Program Quarterly Progress Report", for January - March 1981, ORNL/TM-7822, pp. 76-83.

3. Letter, R. M. Bernero (NRC) to A. L. Lotts (ORNL), "Report on Pressurized Thermal Shock," dated May 11, 1981.

2.0 OVERCOOLING TRANSIENTS IDENTIFIED AS SAFETY CONCERNS

There are three basic mechanisms for rapidly cooling the primary coolant system: depressurization of the primary or secondary system, injection of cold fluid, and rapid removal of energy through the steam generator. Four general classes of transients can be identified as encompassing one or more of these cooling mechanisms:

- Large-Break Loss-of-Coolant Accident (LBLOCA)
- Small-Break Loss-of-Coolant Accident (SBLOCA)
- Main Steam Line Break (MSLB)
- Runaway Feedwater Transient (RFT)

The severity and probability of occurrence of each of these transients is dependent on plant-specific characteristics.

The LBLOCA produces primary fluid temperature temporal derivatives on the order of 36,000°F/hr, arresting at a base temperature of ~350°F. To this system is injected 40-85°F high pressure injection system (HPIS) fluid and 90°F core flood tank (CFT) fluid, which results in rapid chilling of the fluid next to the RV and causes an effectively conduction-limited temperature transient in the vessel wall.

The SBLOCA, in contrast, produces order of magnitude lower primary fluid temperature temporal derivatives than the LBLOCA, generally less than 2200°F/hr, depending on the size of the break. Also, depending on break size, the CFT system may actuate in addition to the HPIS. A critical difference between SBLOCA and LBLOCA is that the HPIS can repressurize the system for many break sizes.

The MSLB usually produces primary fluid temperature temporal derivatives that lie between those of the LBLOCA and SBLOCA. These decreasing temperatures result from the rapid primary system energy removal produced by flashing of the fluid on the secondary side of the steam generator. The lowest primary fluid temperature achievable in this transient is determined by the performance of the steam generator feed train, HPIS, and CFT.

The RFT is essentially a variant of the MSLB, but without the initial rapid steam generator secondary-side blowdown and the resultant rapid removal of energy from the primary system. The primary system temperature temporal derivatives for the RFT are usually the lowest among the four classes of transients. The progress of the RFT is totally controlled by the performance of the steam generator feed train.

3.0 SEVERITY OF THE THREAT

3.1 Probability of Occurrence of Initiating Events

Ultimately, the probability of a thermal-stress-induced challenge to the reactor pressure vessel is dependent on the frequency of requisite initiating events. However, the concern to this study is not the probability of individual initiating events themselves, but rather the total probability that the thermal-hydraulic transients resulting from the initiating events produce pressure and temperature conditions which approach the structural limitations of the RV. This total probability can be viewed as the multiplicative combination of three probabilities: (1) the probability of an initiating event, (2) the probability that the control and safety systems fail to respond to the transient in an appropriate manner to protect the vessel, and (3) the probability that the reactor operator fails to diagnose the exact nature of the transient and therefore fails to take appropriate action or possibly takes action which actually exacerbates the transient.

As noted previously, four transients were identified as possible thermal shock initiators: a small-break LOCA, a large-break LOCA, a main steam line break, and a runaway feedwater transient. Due to limited time and resources, a detailed characterization of the various factors (i.e., initiating events and system/operator responses) for each transient has not yet been performed. An estimate of the probability of each initiating event which could be a precursor to conditions having the potential for thermal shock to the RV was made.

	Probability Per Reactor Year	
	Estimate	Range
SBLOCA[a]	3×10^{-4}	3×10^{-5} to 3×10^{-3}
LBLOCA[a]	1×10^{-4}	1×10^{-5} to 1×10^{-3}
MSLB[a]	5×10^{-6}	1×10^{-6} to 1×10^{-4}
RFT	1.0	0.1 ~~1.0~~ To 1.0

The RFT event is the most complex with a very high probability for the initial event but requiring failure in order to produce thermal shock. This is very plant specific, and for Oconee-1 it appears that multiple independent failures are required (see Section 4.5.4).

[a]Estimated from Reactor Safety Study, WASH-1400.

Table 3-1. Summary of Pressurized Thermal Shock Evaluation Mechanistic Results

INITIATING EVENTS

SUBSEQUENT EVENTS	Large-Break LOCA (LBLOCA)	Small-Break LOCA (SBLOCA)	Main Steamline Break (MSLB)	Runaway Feedwater Transient (RFT)
Thermal-Hydraulic Information Sources	TRAC simulation (Westinghouse Plant)	(a) Rancho Seco (actual plant transient) (b) TRAC simulation	(a) TRAC simulation (b) IRT simulation	(a) Rancho Seco (b) IRT simulation
Operator Actions Taken or Assumed	None	(a) Rancho Seco operator overrode automatic trip of main feedwater pumps (b) TRAC: initiates aux. feed water at 30 s. Main feedwater ramped at 40 s.	(a) TRAC: initiates auxiliary feedwater at 30 s. Main feedwater ramped at 40 s. (b) IRT: none	(a) See SBLOCA (b) None
Thermal-Hydraulic Indications or Predictions	No repressurization of primary coolant system	(a) Rancho Seco: - Repressurization - Tmin = 280°F (b) TRAC: analysis terminated prematurely	(a) TRAC: - No repressurization - Tmin = 350°F (b) IRT: - Repressurization - Tmin < 150°F	(a) See SBLOCA (b) - Repressurization - Tmin < 150°F
Vessel Fracture Mechanics Predictions	Crack initiation and arrest (no vessel failure) at 20 EFPY	(a) Vessel fails at 20 EFPY (b) Insufficient information for analysis	(a) TRAC: Analysis not yet available (b) IRT: Vessel fails at 4 EFPY	(a) see SBLOCA (b) IRT: Vessel fails at 3 EFPY
Limitations and Concerns	No repressurization, so of secondary concern	(a) Ranch Seco: - Pressure & temperature data not entirely adequate (b) TRAC: - Press/temp. data incomplete	(a) TRAC: - Mild case - Feedtrain as tables (b) IRT: - Assumes feedwater control failure - Feedtrain as tables	(a) See SBLOCA (b) - Assumes feedwater control failure - Feedtrain treated with tables

3.2 Pressure Vessel Integrity

For the purpose of these studies the integrity of the pressure vessel was considered to be jeopardized if the fracture mechanics analysis indicated that an inner surface flaw would propagate through the vessel wall as a result of an overcooling accident. Four specific overcooling accidents for which fracture-mechanics analyses were performed are the large-break LOCA, Rancho Seco (1978), turbine trip with stuck-open bypass valves, and main steam line break. Results of the preliminary analyses indicate that the vessel will not fail as a result of the LBLOCA, but failure was predicted for the other three accidents. The calculated threshold times for failure were 20, 3 and 4 EFPYs, respectively, based on a fluence rate of 0.046×10^{19} neutrons/cm^2/EFPY, which is similar to that for B&W plants.

The size of the break that results from propagation of the flaw through the wall is of utmost importance since it is a factor in determining whether the vessel will be able to retain sufficient water to cool the core. Because cooling temperatures at some, if not all, locations in the primary system are expected to be well above 212°F at the time of predicted failure (excludes LBLOCA), there is a large amount of stored energy that will be released, and thus a potential exists for a rather large opening in the vessel wall. A more quantitative assessment of the problem awaits completion of detailed studies.

4.0 PLANT CONTROL AND OPERATIONS

4.1 Introduction

Plant control systems and operator actions were reviewed to define system setpoints and capacities relevant to pressurized thermal shock transients. Potential feedwater control failures and operator actions were investigated and the control system and operator actions which took place in the Rancho Seco overcooling transient were reviewed. These data were used to evaluate control system response assumptions employed in the thermal-hydraulic analyses available to us and to develop conclusions and recommendations concerning control system modeling for pressurized thermal shock thermal-hydraulic analysis, operator actions, and potential problem areas.

4.2 Reactor Protection System

The Reactor Protection System (RPS) is a safety-grade system designed to trip the reactor according to the values of a variety of input parameters, and thereby to protect both the core from fuel rod cladding damage and the reactor coolant system from overpressurization.[1] Reactor trip directly influences main feedwater control through the Integrated Control System[2] (ICS) unit load control. The neutron power signal obtained from the RPS can also modify main feedwater demand if its mismatch with the ICS reactor demand level exceeds a set tolerance.

Following a reactor trip, the heat generated by the reactor is determined by the shutdown rate. In order for the remainder of the unit to "follow" the reactor, the unit load demand (and hence the total feedwater demand) will track the actual megawatts generated at a maximum rate of 20% per minute. For transients involving initial depressurization, the RPS will trip the reactor at a low pressure set point of 1925 psi. For transients initiated by a turbine trip, the reactor will be tripped at the start of the transient.

4.3 High Pressure Injection

4.3.1 System Description

The High Pressure Injection System (HPIS) injects water into the four reactor vessel inlet pipes upon actuation of appropriate trips in the engineered safety features system. The HPIS comprises three high-pressure pumps; the flow can be controlled manually and the pumps can be aligned manually in several different ways.[3] Normal HPIS actuation will inject full flow from two of the three pumps, with suction taken from the borated water storage tank (BWST).

4.3.2 Capacities[1]

The HPIS pump characteristic performance curve is shown in Fig. 4-1. The BWST has a volume of 388×10^3 gallons. The HPI valves will be fully open within 14 s from an actuation signal, and the pumps will be up to speed within 6 s.

4.3.3. Set Points[1]

The HPIS will actuate when the reactor coolant system pressure drops to 1500 psi.

4.4 Low Pressure Injection

4.4.1 System Description

The Low Pressure Injection System (LPIS) injects water into the reactor vessel downcomer through two pipes located on opposite sides of the core and at ninety degrees from the reactor vessel outlet nozzles. Low pressure injection is provided by three low-pressure pumps (operated in parallel) and two accumulators.* Pumps are normally aligned to draw from the BWST, but can be manually transferred to take suction from the reactor building sump. The pump flowrate can be controlled manually.

4.4.2 Capacities[1]

The LPIS pump characteristic performance curve is shown in Fig. 4-2. The LPI valves will be fully open within 15 s after actuation, and the pumps will be up to speed within 8 s.

As stated previously, the BWST has a capacity of 388×10^3 gallons. When considering the total inventory of borated water available to the LPIS, it must be noted that the containment spray system, if actuated, will also draw from the BWST. The accumulators have a combined capacity of 21×10^3 gallons.

4.4.3 Set Points[1]

The accumulators will discharge water into the reactor vessel when the pressure falls below 600 psi, and the LPIS pumps will be actuated when the primary pressure falls below 200 psi.

4.5 Main Feedwater Control

4.5.1 System Description

Main feedwater control is one function of the ICS. A feedwater demand signal is developed, based on unit load demand but also ratioed and

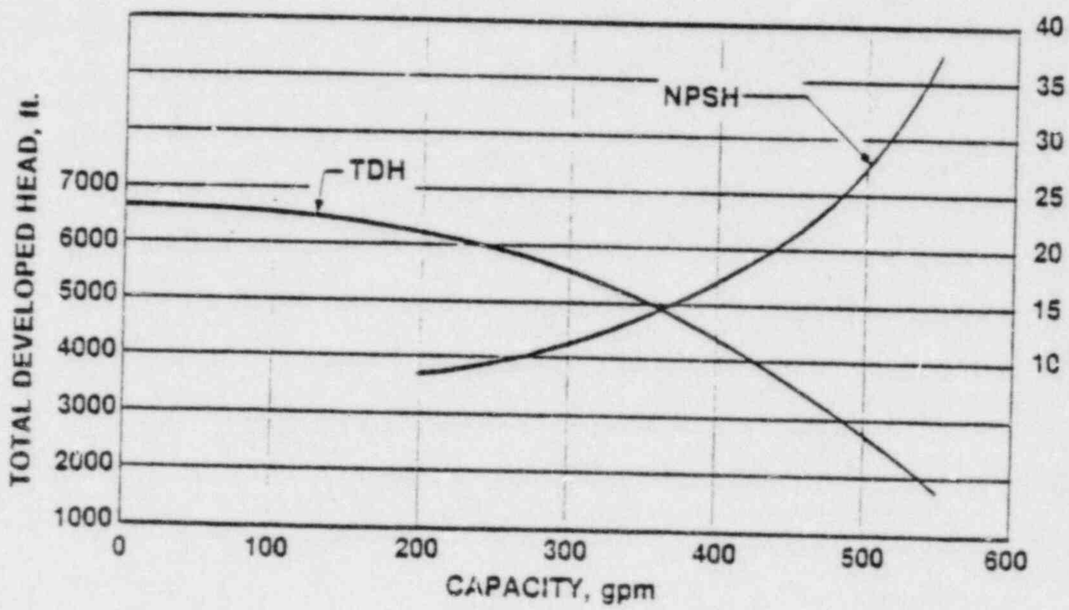

Fig. 4-1. High Pressure Injection Pump Characteristics (From Oconee FSAR).

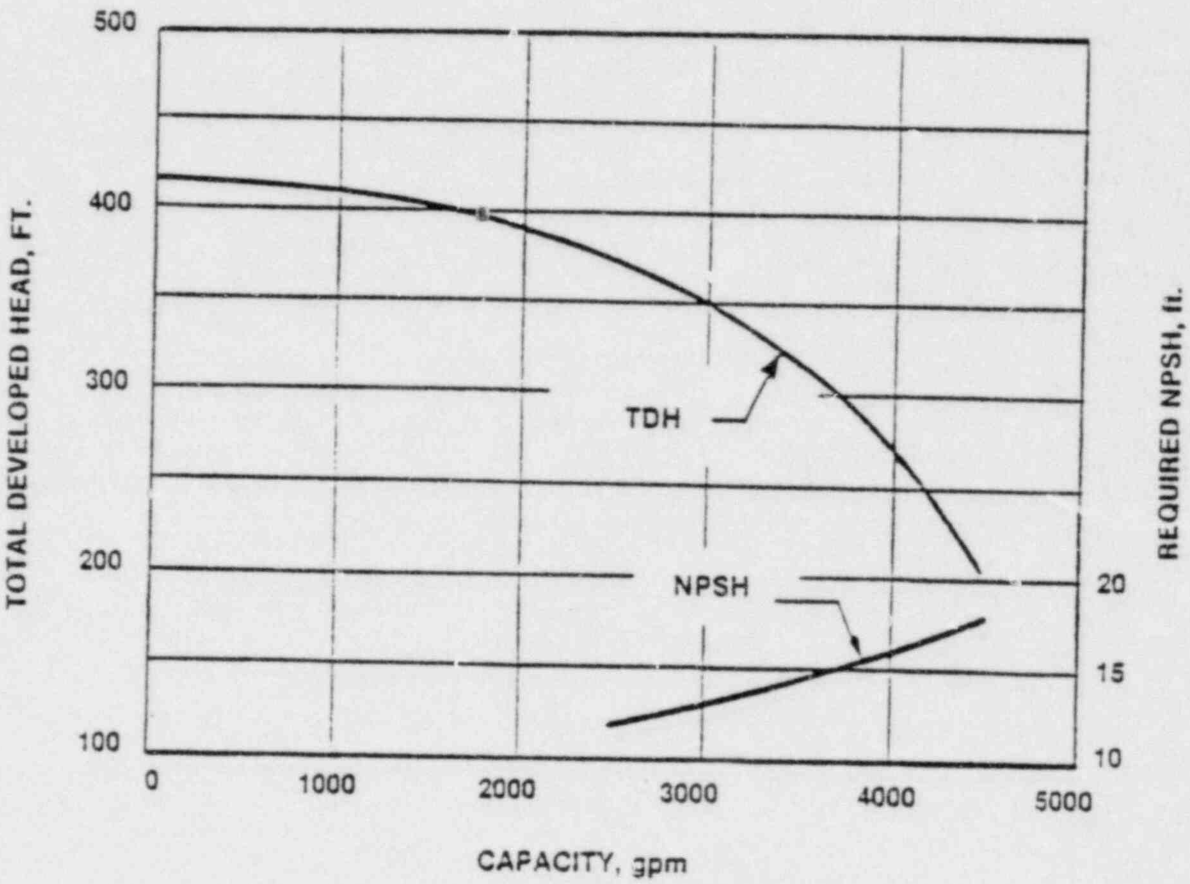

Fig. 4-2. Low Pressure Injection Pump Characteristics (From Oconee FSAR).

limited by a number of feedback functions. The loop demand signal is compared to the measured feedwater flowrate so as to regulate the main feedwater control valve. The total corrected feedwater demand signal is modified to control the feedwater pump speed, and so to meet the feedwater demand and to maintain constant pressure drop across the main control valves. Steam generator low-level and high-level limits intercept the feedwater demand to prevent underfill and overfill conditions.

The main feedwater pumps are supplied water from the condenser hotwell, the surge tank, and the condensate storage tank through three condensate booster pumps and three hotwell pumps.

A variety of feedback and trip functions affect the main feedwater control. Manual control of all feedwater pumps and valves is also possible.

4.5.2 Capacities[1]

The full main feedwater flow is 6539×10^3 lbm/hr from the hotwell per steam generator. The maximum water inventory in the feedwater system is 295×10^3 gallons.

4.5.3 Set Points

- The total feedwater demand will run back at a maximum rate of 20% per minute to track generated megawatts following a reactor trip.[2]
- Operators are presently required to trip the reactor coolant pumps following actuation of the engineered safety features system. Tripping the reactor coolant pumps will, in turn, cause the feedwater demand to run back to a maximum value of 20% at a rate of 50% per minute.[2]
- The Oconee-1 unit has a steam generator high-level limit that will trip the main feedwater pumps if the steam generator is filled to this level.[8] (This trip may not be present on other B&W units).
- The Oconee-1 unit will also trip both main feedwater pumps if there is a loss of ICS power.[8] (This trip may not be present on other B&W units.)
- All hotwell pumps will be tripped[8] when the hotwell level reaches "emergency low."
- All condensate pumps will be tripped when the condensate booster pump suction header pressure decreases below 43 psig.[8]

4.5.4 ICS Failure Modes and Effects

The ICS is a complex, nonsafety-grade control system. A failure modes and effects analysis (FMEA) review was therefore performed to

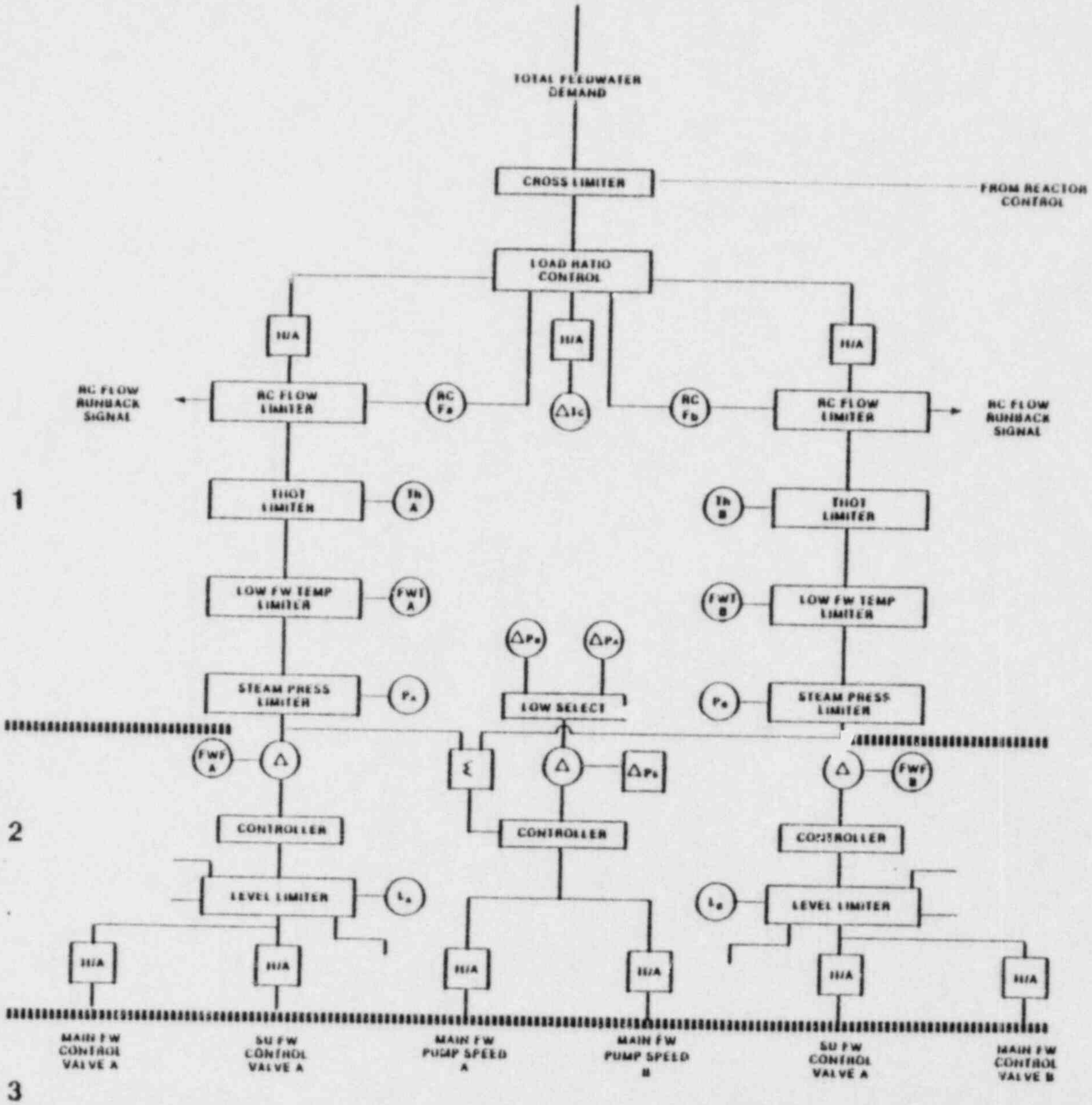

Fig. 4-3. ICS General Area Breakdown.

investigate potential failures in the ICS that might lead to excessive feedwater. This review divided the ICS into three general areas, as shown in Fig. 4-3. The first area constitutes higher levels in the ICS, where significant feedback and feedwater flow limiting actions will be effected from a variety of other process signals. The second area is characterized by limited ICS feedback, but manual control of the feedwater system is still possible. The third area encompasses failures below all control points.

The results of this analysis, which are summarized in Table 4-1, show that single control failures below the manual control points always leave one or more alternate means by which the feedwater may be manually controlled. Manual control is required, in general, to assure proper feedwater control. ICS failures in region 2 will limit feedwater flow to the steam generator high-level limit. In the case of Oconee-1, if the ICS high-level limit does not act, a separate high-level signal will trip the main feedwater pumps. Feedwater flow will be limited for ICS failures in region 1 by both the high-level limits and by feedback from other parameters. It should be noted that without the steam generator high-level trip for the feedwater pumps, failure of the startup level signal to a "low" condition can result in unlimited overfeed to the steam generator.

ICS failure or power failure will generally lead to a loss of feedwater, due to a zero demand speed signal being presented to the main feedwater pumps. Oconee-1 also has a trip to stop the main feedwater pumps on loss of ICS power. The presence of feedwater control for a loss of ICS power condition is known to be highly plant specific.

4.6 Rancho Seco Transient

A significant overcooling transient occurred on March 20, 1978 at the Rancho Seco reactor following a failure of power supplies that fed both control room indicators and the ICS. The initial plant response was a loss of feedwater transient combined with incorrect indications presented to the operators. Auxiliary feedwater was also supplied to one steam generator through an ICS control path (no longer present in B&W units). As a result, the operator was presented with an indication of 0% feedwater demand for one loop and 100% feedwater demand for the other. His response was to manually remove the main feedwater pump trips and so restore main feedwater. Upon restoration of the instrumentation power, it was discovered that the reactor had been overcooled and corrective actions were taken.[3] The instrumentation and control system power supplies have since been upgraded.

The Rancho Seco incident clearly demonstrates that significant overcooling transients can be induced by operator action. In this case, the operator's actions were the result of unsuspected and

Table 4-1. FMEA Review for Excessive Feedwater Failures and Control

	Failure Mode	Long-Term Control
Below ICS Manual Station		
Feedwater Pump(s)	Fails to full speed	ICS will close FW valve to maintain proper flow. Valves can also be controlled manually.
Main Feedwater Control Valve	Valve Fails Open	Block valve and pump speed can be controlled manually.
Main Feedwater Block Valve	Valve Fails Open	No effect except for leakage around control valve (when closed). Pump speed and control valve can be controlled manually.
Startup Feedwater Control Valve	Valve Fails Open	Startup feedwater block valves can be closed manually.
Startup Feedwater Block Valve	Valve Fails Open	Two block valves available. Startup feedwater control valve can also be closed manually.
Below Other Parameter Feedback		
Feedwater Flow	Fail Low	Control valves open; control valve and block valve can be controlled manually. (Fills steam generator to high-level limit.)
Startup Level	Fails Low	Feedwater valves fail open independent of feedwater demand. Manual control of valves and pump speed possible. Oconee-1 has separate high-level feedwater pump trip.
Loop Feedwater Demand Error	Fails in direction to open valve	Valve can be controlled manually. (Fills steam generator to high-level limit.)

Table 4-1. (Continued)

	Failure Mode	Long-Term Control
Above Other Parameter Feedback		
Loop or Total Feedwater Demand	Fails in direction to increase demand	Valves and pump can be controlled manually. (Cross limits, BTU limits, and level limits limit increase and control at high level.)
Other		
ICS AC Power	Loss of AC Power to ICS	Flow control valves freeze. Feedwater pump goes to low-speed limit. Oconee-1 has feedwater pump trip on loss of ICS power.

widespread information failure. It should be noted that as a result of this and other power-supply-failure-induced control system transients, all B&W units have since been required to review their system power supplies and make modifications and develop procedures as necessary to reduce the probability of such events and to provide operator guidance for controlling the unit.

4.7 Operator Actions

Our review of the ICS failure modes and effects analysis (FMEA) indicated that a few single or double control failures can lead to uncontrolled feedwater supply, but that the majority of potential failures have feedback paths that will act to reduce feedwater automatically. It is concluded that there are a larger number of ways that excessive feedwater can result from operator errors of commission than from errors of omission.

Oconee-1 event sequences for small steam line break and excessive feedwater were reviewed for indicated operator actions. These event sequences assumed multiple system failures and failure to control different systems properly along the event paths. Potential severe consequences were identified where key problems remained uncorrected. These event sequences show the necessity for correct operator actions along a number of paths.[5]

4.8 Review of Thermal-Hydraulic Calculations

Available thermal-hydraulic analyses were reviewed for correctness of assumed control system response. These analyses were found to include operation of the safety systems as designed but to employ rather arbitrary assumptions about feedwater system operation.

In particular, the TRAC calculations for MSLB and SBLOCA assumed that the main feedwater continues to supply 100% flowrate and is then reduced to zero by operator action over a two-minute period starting 40 s into the transient. Emergency feedwater is assumed to be initiated at 30 s to the remaining intact steam generator for the MSLB and to both steam generators for SBLOCA. In the absence of such operator actions, automatic control system actions would perform the same functions, but the effect of the possible difference in timing on the thermal-hydraulic transient is not known at this time.

The IRT calculations for MSLB and RFT assume that the main feedwater continues to supply 100% flowrate to one steam generator for the duration of the feedwater supply. Owing to the wide variety of feedwater control feedback and trip functions present, as described in Sect. 4.5., achievement of 100% flow rate during this tranisent is not possible; the actual achievable flow rate and its effect on vessel integrity are not known, but the lower flow rate would result in a less severe transient.

Therefore, the IRT and TRAC feedwater flow assumptions in both the IRT and TRAC can be considered to be approximately bounding cases for excessive feedwater supply. Another perspective would be to view the cases as representing two events of different probability. The TRAC case assumes correct operation of the feedwater control system, whether by ICS or the reactor operator. The IRT case corresponds to an overt operator error (manually supplying full feedwater flow) or a multiple control system failure with lack of corrective operator action. On this basis we consider the IRT transient to be less probable by a factor of 10^{-3} to 10^{-6}.

4.9 Summary

The plant control and operations review identifies a variety of potential failures that could possibly result in excessive feedwater supply. Most of these failures have feedback or trips that can be expected to reduce the feedwater in a fairly short period of time. The operator can also take action to terminate main feedwater for all single and double ICS failures identified.

The thermal-hydraulic calculations reviewed employed somewhat arbitrary but approximately bounding feedwater response assumptions. The assumptions used in the IRT analyses, in particular, are considered to be conservative, with regard to the severity of the transient.

More feedwater supply calculations would require modeling the feedwater demand runbacks, limits, and feedwater system trip points that were described in Sect. 4.5. Using a model of this type, it would be necessary to investigate several of the potential control system failures to identify the worst credible case. The probability of excessive feedwater is likely to be dominated by the probability of overt operator error, since the control system failure requires two independent failures plus lack of operator corrective action.

REFERENCES — CHAPTER 4

1. Oconee-1 Final Safety Analysis Report.

2. Oconee-1 Instruction Book for Integrated Control and Non-Nuclear Instrumentation Systems.

3. Sacramento Municipal Utility District Followup Report to Reportable Occurrence 78-1, Re Docket No. 50-312, Operating License DPR-54, dated March 31, 1978.

4. Oconee Unit 3 Piping Drawings: PO-101-A, B High-Pressure Injection; PO-102-A Low-Pressure Injection; PO-121-A-3, B-3A, B-3B, D-3 Condensate, Feedwater, Emergency Feedwater Systems (from Duke Power Company).

5. Oconee Small Steamline Break and Excessive Main Feedwater Event Trees and Safety Sequence Diagrams (from Duke Power Company).

6. Transient Response of Babcock & Wilcox - Designed Reactors, NUREG-0667 (May 1980).

7. Integrated Control System Reliability Analysis, BAW-1564.

8. Duke Power Company, personal communication.

9. "Loss of Non-Class-1-E Instrumentation and Control Power Systems Bus During Operation," USNRC IE Bulletin 79-27.

5.0 THERMAL-HYDRAULICS

5.1 Literature Search for Accumulated Experience

Data Bases. Thirteen dat bases

- BHRA
- CIM
- COMPENDEX
- CONF
- EDB
- EIA
- FEDEX
- ISMEC
- NSA
- NSC
- NTIS
- RSI
- WELDASEARCH

were searched for thermal-hydraulic system data relevant to thermal shock tranisents. Capsule descriptions of the data bases are given in Appendix A. Approximately 600 thermal-shock-related entries were found; however, the majority of the entries were LBLOCA emergency core cooling injection studies and generally contained no system thermal-hydraulic information. There were also a significant number of licensee event reports (LERs); however, they focus on the incident precursors, not the transient data, and so are of limited usefullness to this study.

Oconee Licensee Event Reports. Five LERs of interest, covering four events, were found for the Oconee Nuclear Power Station (see Appendix B). The four events were distributed as one each for Units 1 and 2, and two for Unit 3. Three events were in the class of steam generator overfeeds (RFT), and the other was an open power-operated relief valve (SBLOCA). None had major consequences, since the operators took timely action.

Specific Documents. Twenty specific documents, as listed in Appendix C, were also reviewed. The first seven of these references were particularly interesting since they contain system thermal-hydraulic data.

5.2 Thermal Shock Plant Transient Data

The major source of actual plant data for transient overcooling is the March 20, 1978 Rancho Seco event (Appendix C., Refs. 1 and 2). The pressure and temperature data employed by ORNL as input for fracture mechanics calculations (Ref. 2) are shown in Figs. 5-1 and 5-2, respectively. The pressure surges contained in the original data (Ref. 1) were removed for simplicity (they may or may not be "real").

Owing to the locations of the inlet temperature measuring instruments, which are placed in wells at the suction side of the reactor coolant pumps and are therefore upstream of the HPIS injection, the use of these actual plant data as the RV forcing functions foi a B&W plant introduces some uncertainty. The temperature of the fluid at the RV inlet nozzle might therefore be expected to be lower than measured by the instrumentation, unless two-phase thermal equilibrium flow exists.

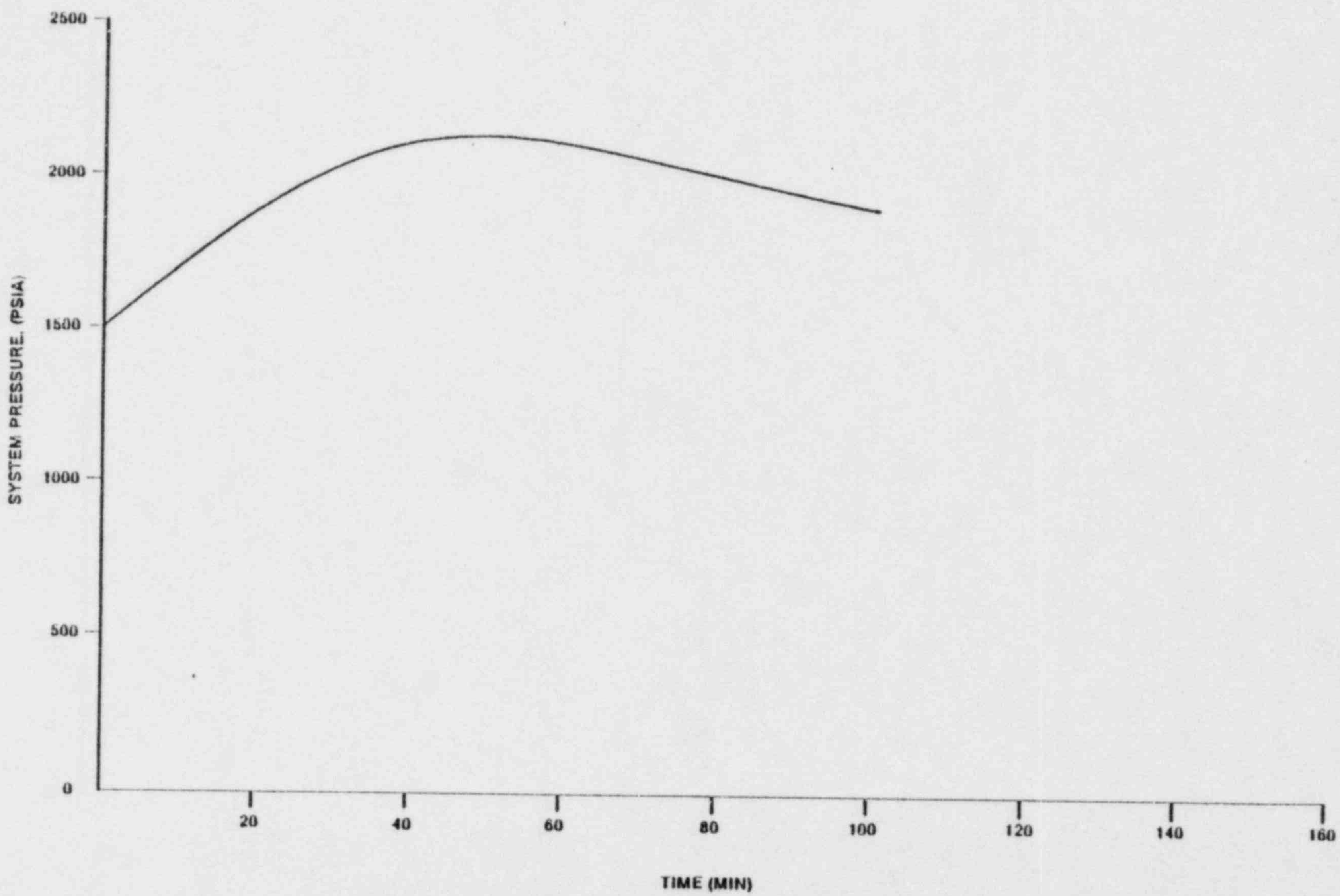

Fig. 5-1. Rancho Seco Pressure Data, as used by ORNL.

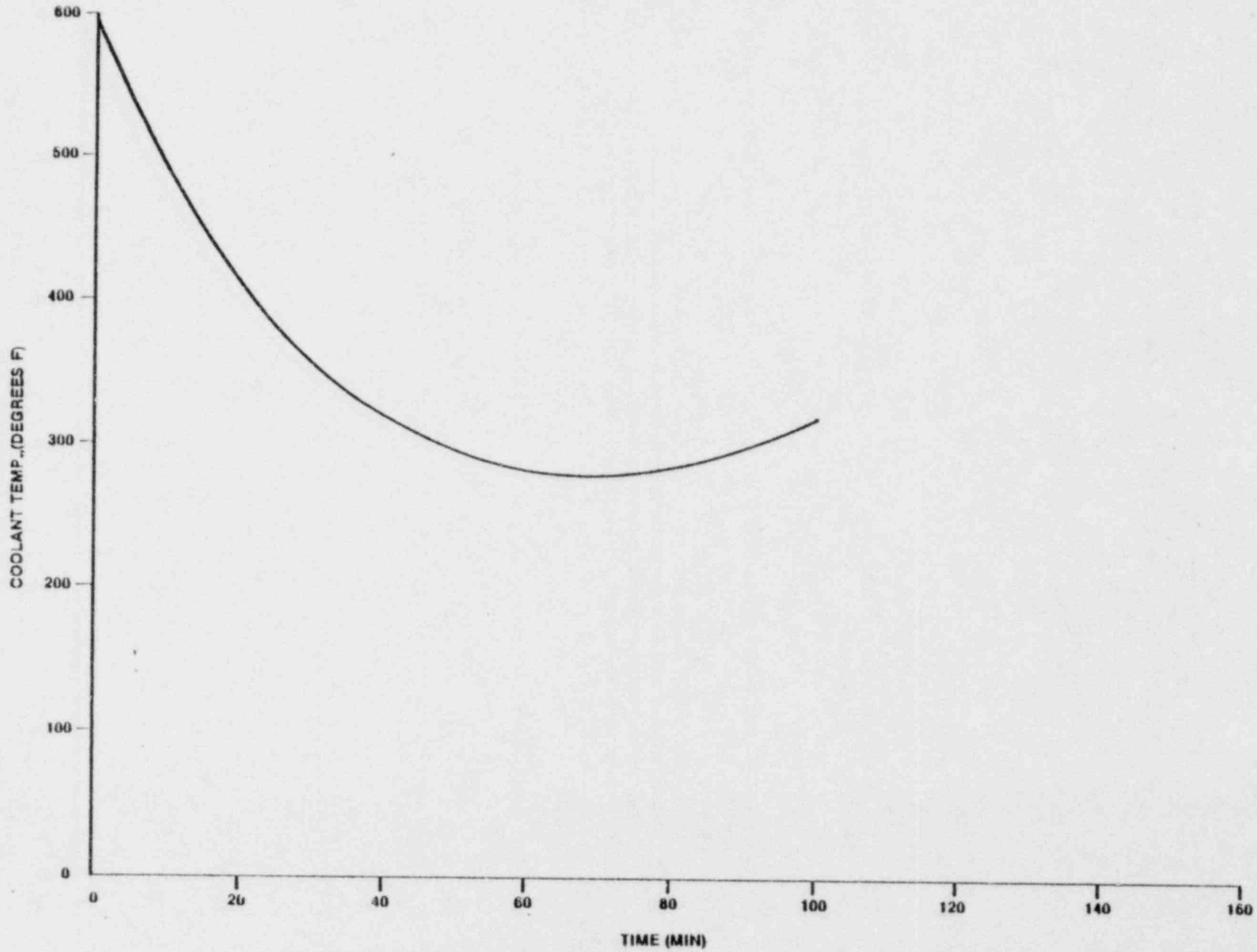

Fig. 5-2. Rancho Seco Temperature Data, as used by ORNL.

On the other hand, for transients in which the vent valves open (not likely for the Rancho Seco event, since the reactor coolant pumps are running, but quite likely if the pumps are shut off), mixing in the downcomer could be significant and the fluid temperature at the RV wall could be higher than the measurement (again, barring two-phase thermal equilibrium flow). Due to the unknown nature of these competing effects, the RV wall temperature forcing function is not easily derived from standard instrumentation in B&W plants during overcooling events.

5.3 Overcooling Simulations

5.3.1 Results of Analyses

To our knowledge, two computer codes, IRT and TRAC, have been used to predict the thermal-hydraulic characteristics of overcooling transients for Oconee-type plants. The pressure and temperature predictions for five scenarios analyzed to date are shown in Figs. 5-3 and 5-4.

Note that primary system repressurization is predicted for all cases except case 4, which corresponds to MSLB as simulated with TRAC, and case 5, for which the transient predictions are incomplete. Repressurization does not occur in case 4 because of an assumption of thermal equilibrium made in modeling the pressurizer. The initial depressurization is similar in all cases, except that IRT does not predict so sudden a primary system contraction as TRAC. The lower worth assumed for the control rods in case 2 is responsible for the quick return of system pressure, as compared to case 1.

Figure 5-4 shows the degree of primary system overcooling to be similar for cases 1 and 3. The lower rod worth of case 2 is again evident in the rate of cooldown. Case 4, TRAC MSLB, is the only one which does not show great overcooling; this difference is attributable to assumed operator termination of main feedwater flow at 40 s into the transient and the use of emergency feedwater to the intact steam generator.

5.3.2 Limitations

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, as discussed below.

Feed Train. Owing to the multiplicity of heaters and pumps and the various input conditions and feedbacks present, it is not easy to calculate the steam generator secondary-side inlet conditions. Accordingly, they are supplied by look-up tables in both IRT and TRAC, and the values entered are the result of simplifying assumptions. Moreover, since this tabular input is fixed for the duration of the transient, the course of the RFT and the latter stages of the MSLB cases are almost completely determined by the values entered initially in the look-up tables.

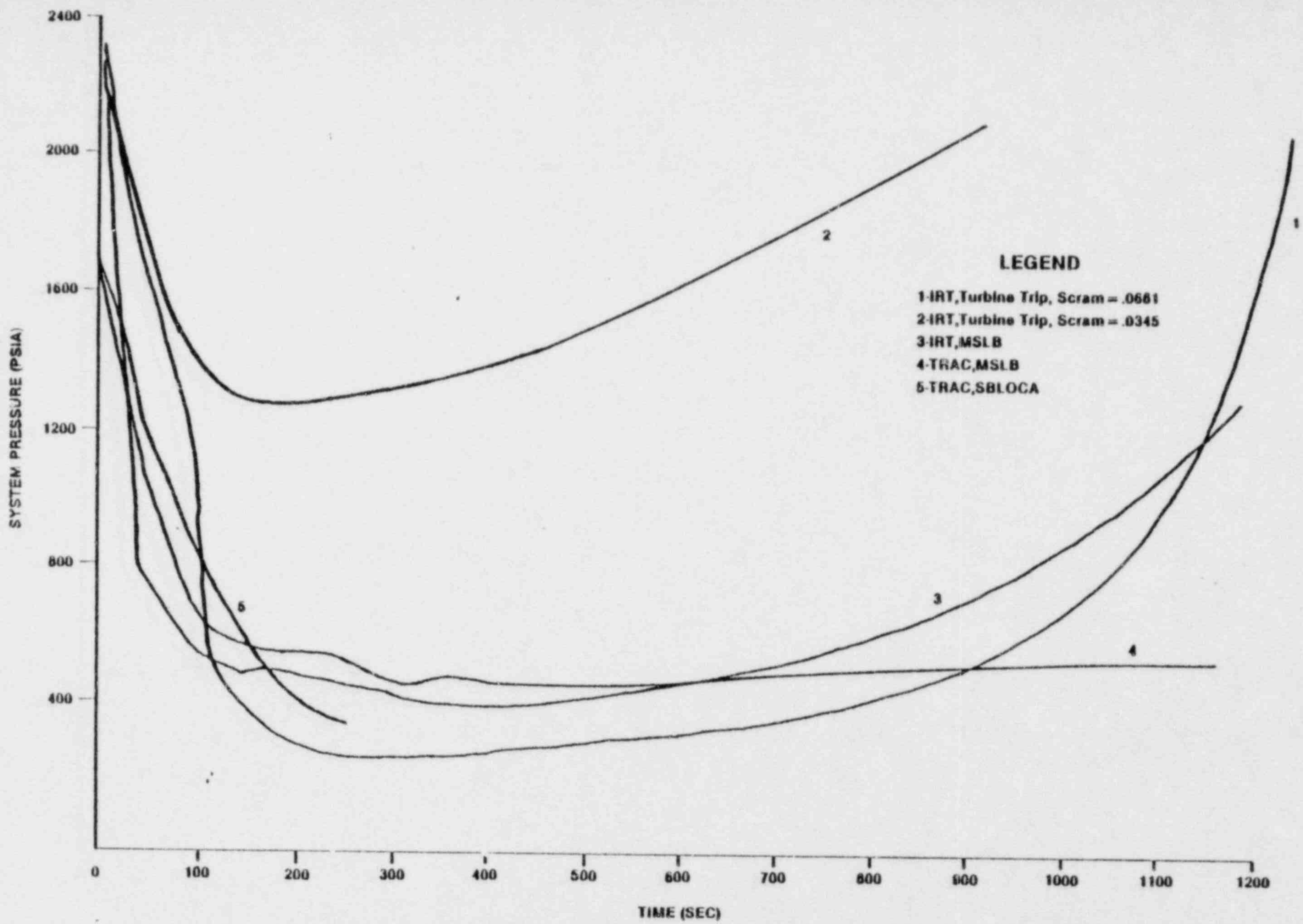

Fig. 5-3. Composite Pressure Profiles from Simulations.

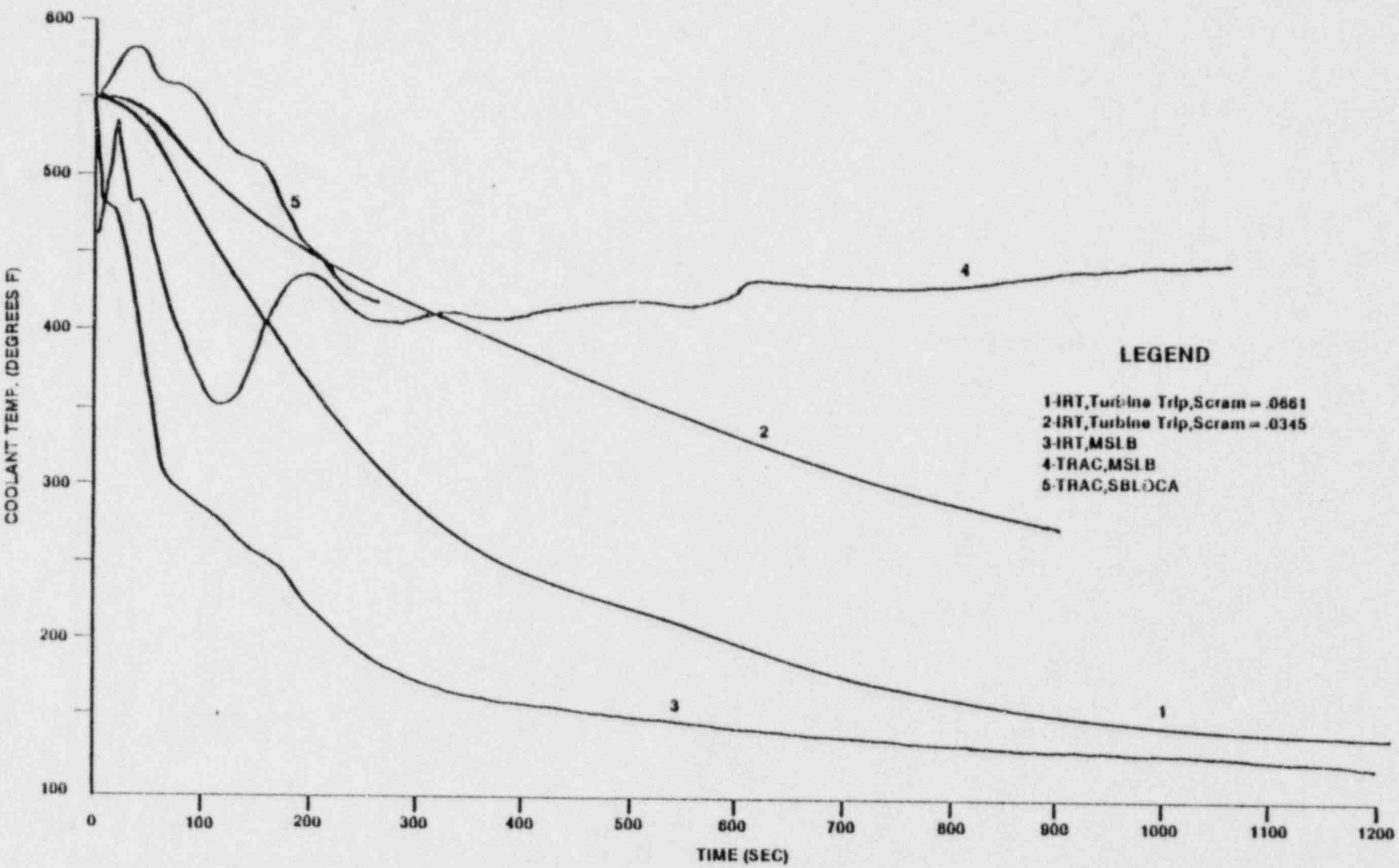

Fig. 5-4. Composite Temperature Profiles from Simu'ations.

Fluid Mixing. It is obvious that the degree of mixing between the HPIS and primary-system fluids will have a marked effect on the RV wall temperature in the downcomer region. Further, as shown in Appendix C, Ref. 3, for cases where the vent valves open, the degree of fluid mixing assumed to occur in the upper downcomer completely determines the severity of these transients.

Neither TRAC nor IRT contains models which are suitable for analyzing these special cases. IRT has only equilibrium capability for the HPIS and vent valve mixing, whereas TRAC has non-equilibrium capability. It is our understanding that the cases modeled thus far do not predict vent valve opening, so some basic code differences may not be manifested in the simulation results.

Control Systems. The performance of the control system was assumed and specified before the cases were run; the assumptions were rather arbitrary and, in most cases, quite conservative. Direct feedback control system modeling for these transients is not currently possible in either TRAC or IRT.

Repressurization. Repressurization is a key phenomenon, both as to its eventual occurrence and the specific time at which it occurs within the transient, since this relates to likely operator actions. Only bounding cases have been run thus far, i.e., full equilibrium or noninterchange pressurizer models have been used. Actually, more than the pressurizer is involved here; the upper head and the entire hot leg act as "pressurizers" during these transients and their performance during primary system refill will also affect the rate of repressurization.

Flow Distribution. An ability to calculate flow in the primary system over conditions ranging from full forced flow to natural circulation is required to treat these transients properly. How well the codes perform such calculations has not been determined, although to date the IRT calculations have apparently been "driven" by input and have not utilized a momentum equation solution.

Existence of Two Phases. Owing to void formation at the top of the "candy cane," loss of natural circulation will occur in B&W plants during many overcooling transients. IRT cannot treat this phenomenon correctly, which obviously limits the transients to which this code is applicable.

Primary Metal. The effect of heat transfer to the primary fluid from the primary metal has not been fully evaluated. Such heat transfer has been included in some cases (IRT-RFT, TRAC-MSLB and SBLOCA) and not in others (IRT-MSLB). The effect could be significant in some transients, and so should be included in realistic evaluations.

IRT Cases 1, 2, and 3. These are two cases of turbine trip with open bypass valves (Cases 1 and 2) and a main steam line break (Case 3). In all cases full main feedwater flow is assumed. This is a very conservative assumption. With one steam generator flooded and the intact steam generator isolated, the operator would have to go to extremes to keep the main steam-driven feed pumps running, having lost his primary steam source (the steam generators). The hot well and condensate booster pumps do not have enough head to maintain full flow (at least in Cases 1 and 2). In addition, a multiplicity of ICS failures would have to be assumed to prevent automatic runback and trip of the main feedwater train.

The feedwater temperature was assumed to be a simple ramp down to the hot well temperature, 91°F, over one minute. This is again conservative. The several pumps and heaters and the length of piping will all have considerable capacity to hold the temperature up. In addition, the high-pressure heaters take their steam from the main steam lines and not the turbine, so they will not be isolated by the trip.

The rate at which the cooled primary system fluid is transferred to the pressure vessel is not properly calculated. With the primary pumps tripped and the system depressurized, voiding in the "candy cane" will inhibit natural circulation until the system is refilled by the HPIS. The repressurization by the HPIS is also overpredicted. The non-interchange models used for the upper head and pressurizer result in a steam compression calculation producing the pressure. Since the steam and water are assumed not to interact, the steam "bubble" is compressed at a volume reduction rate equal to the HPIS charging capacity.

TRAC Case 4. This is a main steam line break and is more realistic in its assumptions of feed train performance. The operator is assumed to start auxiliary feedwater at 30 s and terminate main feedwater at 40 s. The ICS would have performed the same functions in the same time frame had the operator not acted. The auxiliary feedwater comes straight from the hot well, so its temperature is more easily determined.

The adequacy of TRAC to treat "candy cane" voiding and consequent loss of natural circulation and, therefore, the transport of cooled fluid to the pressure vessel before repressurization by the HPIS is not known. However, the repressurization rate is known to be too slow. Full equilibrium is assumed in the node above the water level in the pressurizer. This results in condensation of the steam. Therefore, repressurization will not occur until all steam has been condensed and the pressurizer is full; this is unrealistic.

By comparison, case 4 is more realistic and also much milder than the IRT-MSLB, Case 3, as can be seen in Figures 5-3 and 5-4. Under the limitations noted, these two cases could be considered bounding analyses.

5.4 "Best Judgment" Pressure and Temperature Forcing Functions

Owing to the above deficiencies, the available simulated forcing functions must be regarded as approximations to the true functions; the magnitude and sign of the error is not presently known. Whatever their deficiencies, the Rancho Seco data represent a recorded event, not a simulation, and so provide the closest realistic vessel forcing functions currently available.

References, By Legend Number, for Figs. 5-3 and 5-4

1. "Runaway Feedwater After Turbine Trip Report," letter, M. Levine to N. Zuber dated July 2, 1980.

2. Ibid.

3. "Analysis of a Steam Line Break with Primary System Overcooling for a Typical B&W Reactor," letter, R. Cerbone to R. Kryter dated August 14, 1981.

4. "Completion of Scheduled Analysis on Pressurized Thermal Shock Scenarios," letter, S. Fabic to C. Serpan dated June 22, 1981.

5. Ibid.

6. "Parametric Analysis of Rancho Seco Overcooling Accident," letter, R. Cheverton to M. Vagins dated March 3, 1981.

6.0 ESTIMATION OF NEUTRON FLUENCE AT THE REACTOR VESSEL WALL

6.1 Introduction

A realistic evaluation of a postulated "pressurized thermal shock" accident requires a knowledge of the neutron fluence ($E \geq 1.0$ MeV) and its uncertainty throughout the reactor vessel wall. The fluence values must be known at the location of those materials (welds or plates) which are most likely to be affected by the conditions attained during the transient.

Currently the Code of Federal Regulations (10CFR50, Appendices G and H) and Regulatory Guide 1.99 (Ref. 1) require licensees to provide estimates of the neutron fluence in the reactor vessel beltline region as a part of their surveillance programs. The methodologies adopted by different vendors and service laboratories to obtain the fluences can be separated into three parts:

a. Neutron transport calculations
b. Dosimetry measurements
c. Analysis of uncertainties

The techniques for accomplishing parts "a" and "b" do not vary significantly from vendor to vendor. However, the uncertainty analysis involves combining differential and integral dosimetry data, both measured and calculated, in a consistent fashion so as to obtain absolute fluence values (and their uncertainties) in the RV wall as a function of energy. These fluences must be extracted from an analysis of measurements performed at the location of a surveillance capsule, which may be distant from the RV wall. Although considerable work has been expended in developing methodologies [1-5] to achieve part "c", the application to power reactors has just begun.

A preliminary list of uncertainties affecting the calculation and measurement of neutron flux and fluences in LWRs was compiled by the ASTM E10.05.01 Task Group on Uncertainty Analysis. This list, with a few additions, is given in Table 6.1.

6.2 Babcock and Wilcox Methodology

6.2.1 Neutron Transport Calculations

The calculated energy-group fluxes are determined using a discrete ordinates solution of the Boltzmann transport equation. The codes used are ANISN[6] and DOT.[7] Table 6.2 below gives the steps in the B&W transport calculational procedure.

Table 6.1. Uncertainties for Calculation and Dosimetry Measurement Procedures in LWRs

Source of Uncertainty	Estimated Uncertainty (%)
I. Calculational Procedure[a,b]	
A. Source Term	
1. Fuel management - uncertainty in the fuel cycle	10
2. Fuel position	< 5
3. Burnable poison	10
4. Core power distribution (cycle and cycle-to-cycle variation)	30
5. Power/time history (cycle and cycle-to-cycle variation)	10
6. Local power at core periphery vs. total power	
a. axial	20
b. radial and azimuthal	20
7. Control rod position	5
B. Transport	c
1. Flux synthesis (reduction of 3-D to 1-D and 2-D calculations)	c
2. Energy group structure	c
3. Quadrature (S_n and anisotropic scattering P_ℓ)	c
4. Cross sections	c
5. Spatial mesh	c
6. Geometry	c
7. Time-averaging vs. changing core condition	c
8. Extrapolation (lead factor)	c
II. Dosimetry Measurement Procedures[b]	c
A. Nuclear data (reaction rate cross sections, branching rations, etc.)	c
B. Competing reactions	c
C. Photofission corrections	c
D. Flux/time history	c
E. Counting calibration	c

[a]Values listed compiled by ASTM E/0.05.01 Task Group on Uncertainty Analysis.

[b]BAW-1485 discusses several of the sources of uncertainty, but does not specify the effect on fluence estimates at the RV welds.

[c]Currently unavailable.

Table 6.2. B&W Method for Obtaining Neutron Fluences

a.	Obtain a pin-to-pin, time-averaged power distribution
b.	Obtain P_3 and P_1 22-group CASK cross section sets for 1-D and 2-D calculations, respectively
c.	Perform a 1-D, P_3, S_6 discrete ordinates transport calculation
d.	Perform a 1-D, P_1, S_6 discrete ordinates transport calculation
e.	Obtain a P_3/P_1 correction factor from the 1-D calculations to apply to a 2-D calculation
f.	Perform a 2-D, x-y calculation with the surveillance capsule
g.	Perform a 2-D, x-y calculation without the surveillance capsule
h.	Obtain a capsule perturbation factor from the 2-D calculations to correct the measured activity or calculated fluxes
i.	Perform a 2-D, P_1, r-θ calculation
j.	Perform an axial 2-D, P_1, r-z calculation
k.	Obtain synthesized 3-D group fluxes in the reactor vessel
l.	Correct estimates of the group fluxes, based on the P_3/P_1 and capsule perturbation factors

6.2.2 B&W Surveillance Dosimetry Measurements

The surveillance program for Oconee-1 comprises eight surveillance capsules designed to monitor the effects of neutron and thermal environment on the materials of the reactor pressure vessel core region. The capsules, which were inserted before initial plant startup, are positioned between the thermal shield and the RV wall.

Six of the capsules, placed two in each holder tube, are positioned near the expected peak axial and aximuthal neutron flux. The remaining two capsules (designed to monitor thermal aging) are placed in an area of essentially zero neutron flux.

Capsule OC1-F was removed during the first refueling shutdown of Unit 1, and capsule OC1-E was removed during the second refueling shutdown.

Four activation detectors with reaction thresholds in the energy range of interest were placed in each surveillance capsule. The properties of interest for the detectors are given in Table 6.3, and the results of the averaged measured reaction rates are given in column 3 of Table 6.4.

Table 6.3. Surveillance Capsule Detector Data

Detector	Threshold Energy (MeV)	Product Half-life
$^{59}Co(n,\gamma)^{60}Co$	Thermal	5.26 y
$^{237}Np(n,f)^{137}Cs$	0.5	30 y
$^{238}U(n,f)^{137}Cs$	1.5	30 y
$^{54}Fe(n,p)^{54}Mn$	2.0	313 d
$^{58}Ni(n,p)^{58}Co$	2.5	71.3 d

Table 6.4. Comparison of Calculated to Experimental Reaction Rates for Oconee-1

Capsule	Reaction	Experiment, E (μCi/g)[c]	Calculated, C (μCi/g)	Ratio,[d] E/C
OC1-F[a]	$^{54}Fe(n,p)^{54}Mn$	17.6 ± 0.95	21.0	0.84
	$^{58}Ni(n,p)^{58}Co$	495.0 ± 2	422.0	1.17
	$^{258}U(n,f)^{137}Cs$	1.36 ± 0.21	0.58	2.34
	$^{237}Np(n,f)^{137}Cs$	7.86 ± 0.10	2.90	2.70
OC1-E[b]	$^{54}Fe(n,p)^{54}Mn$	536 ± 62	729.3	0.74
	$^{58}Ni(n,p)^{58}Co$	975 ± 163	1,266.0	0.77
	$^{238}U(n,f)^{137}Cs$	1.94 ± 0.18	1.719	1.13
	$^{237}Np(n,f)^{137}Cs$	9.32 ± 1.18	8.799	1.06

[a] BAW-1421

[b] BAW-1436

[c] ^{54}Mn and ^{58}Co values are given in units of per gram of target for OC1-E and per gram of dosimeter for OC1-F.

[d] Normalization constant

6.2.3 Determination of the Neutron Fluence (E>1 MeV) at the Reactor Vessel

The energy-dependent neutron flux is not directly available from activation detectors because they provide only the integrated flux on the target material as a function of both irradiation time and neutron energy. To obtain an accurate estimate of the average neutron flux incident upon the detector, the following parameters must be known: (1) the operating history of the reactor, (2) the energy response of the given detector, and (3) the neutron energy-group fluxes at the detector location. Two means are available to obtain the desired spectrum: iterative unfolding of experimental foil data and neutron transport methods. Due to a lack of sufficient threshold foil detectors satisfying both the threshold energy and half-life requirements necessary for a surveillance program, the neutron energy spectrum was obtained by the transport method (Sec 6.2.1), instead of by spectrum unfolding.

The calculated spectrum is used in the following equations to obtain the calculated activities used for comparison with the experimental values. The basic equation for the activity, D (in μCi/g) is:

$$D_i = \frac{CN}{A_i} \frac{1}{3.7 \times 10^4} f_i \sum_E \sigma_i(E)\phi(E) \sum_{j=1}^{M} F_j \left(1 - e^{-\lambda_i t_j}\right) e^{-\lambda_i (T-\tau_j)} \qquad (6.1)$$

where

- C = normalizing constant
- N = Avogadro's number
- A_i = atomic weight of target material i
- F_i = either weight fraction of target isotope in the i^{th} material or fission yield of desired isotope
- $\sigma_i(E)$ = group-averaged cross sections for material i
- $\phi(E)$ = group-averaged fluxes calculated by DOT analysis
- F_j = fraction of full power during j^{th} time interval, t_j
- λ_i = decay constant of the i^{th} material
- t_j = interval of power history
- T = sum of total irradiation time, i.e., residual time in reactor and wait time between reactor shutdown and counting
- τ_j = cumulative time from reactor startup to end of j^{th} time period, i.e, $\tau_j = \sum_{k=1}^{j} t_k$

The normalizing constant, C, can be obtained by equating the right hand side of Eq. (6.1) to the measured activity. With C specified, the neutron fluence $\geq$ 1 MeV can be calculated from

$$\phi(E \geq 1.0 \text{ MeV}) = C \sum_{E=1}^{15 \text{ MeV}} \phi(E) \sum_{j=1}^{M} F_j t_j , \qquad (6.2)$$

where M is the number of irradiation time intervals.

The last column of Table 6.4 (Ratio E/C) shows the spread of the normalizing constant as a function of the threshold reaction used in the measurements. BAW-1436 states that the ^{238}U and ^{237}Np reactions from the OC1-F capsule have been deleted in all current evaluations on a basis of inconsistency.

6.3 Results and Uncertainty Analysis

The estimated fluences (E $\geq$ 1 MeV) at the axial welds (Table 6.5) were determined from Tables 6.6 and 8.1 of BAW-1436. The procedures used in obtaining these estimates are given in BAW-1485 (proprietary). The estimated uncertainties in the surveillance capsule analysis are also provided in Sect. 4 (Tables 4-1 and 4-2) and Appendix F of BAW-1485 (proprietary). Fluences at the vessel wall locations may be as high as ±50%.[8]

The procedure outlined in BAW-1485 identified many of the sources of uncertainty stated in Table 6.1 but did not specify all their values.

6.4 Conclusions and Recommendations

The methodology used by B&W is similar to that used by other vendors and service laboratories. One weakness in the methodology is the analysis of uncertainties. This analysis should provide not only estimates for the sources of uncertainty identified in Table 6.1, but a statistically sound technique for <u>combining</u> all the individual estimates to arrive at an overall uncertainty for the fluences at the RV wall. This task requires considerable effort and funding on the part of the vendors, and only recently has work been done in this area.[1-5] Another weakness relates to surveillance dosimetry measurements and the subsequent analysis to obtain fluxes $\geq$ 1 MeV or other suitable neutron exposure parameters. To address this problem,

Table 6.5. Predicted Fast Fluence in Oconee-1 RV at the Axial Welds for 8 EFPY[a]

Beltline Location	Material	Surveillance Locations	Inside of RV Wall	T/4	3T/4	Outside of RV Wall
Upper long. weld	SA-1073	4.95E+18	2.93E+18	1.63E+18	3.70E+17	1.39E+17
Middle long. weld	SA-1493	4.83E+18	2.86E+18	1.59E+18	3.65E+17	1.36E+17
Lower long. weld	SA-1430	6.42E+18	3.80E+18	2.11E+18	4.79E+17	1.81E+17
Peak flux location		7.30E+18	4.32E+18	2.40E+18	5.45E+17	2.07E+17

[a]BAW-1436, Tables 6-6 and 8-1

a pressure vessel benchmark facility for power reactor surveillance dosimetry validation and certification is needed to help identify and reduce uncertainties. It is thought that, with care, an overall uncertainty of ±10-30% for the fluences at the vessel wall should be achievable.

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7.0 PRESSURE VESSEL MATERIAL PROPERTIES

7.1 Material Properties Required for Vessel Integrity Studies

The material properties required for studying vessel integrity can be grouped in accordance with the three types of analyses that must be performed (Table 7.1).

Table 7.1 Reactor Vessel Material Properties Needed for Vessel Integrity Studies

Thermal Analysis	
Thermal conductivity	(k)
Specific heat	(c_p)
Density	(ρ)
Stress Analysis	
Linear thermal coefficient of expansion	(α)
Modulus of elasticity	(E)
Poisson's ratio	(ν)
Yield and ultimate strengths	(σ_y, σ_u)
Fracture-Mechanics Analysis	
Static crack-initiation toughness	(K_{Ic})
Static crack-arrest toughness	(K_{Ia})
Reference nil ductility temperature	(RTNDT)

7.2 Dependence of Material Properties on Chemical Composition, Temperature, and Fast-Neutron Fluence

Generally speaking, all of the material properties in Table 7.1 are functions of material chemical composition, temperature, and fast-neutron fluence and must be treated accordingly in carrying out the vessel integrity studies.

7.2.1 Material Chemical Composition and Heat Treatment

With the exception of a few of the earliest reactor pressure vessels, all belt-line regions of U.S. commercial reactor vessels presently in service were fabricated from the three materials described in Table 7.2. Two additional materials that must be considered are the weld filler material (used to join sections of rolled plate and forging rings) and the vessel cladding (applied by depositing weld metal). The weld filler material is similar to the base material, whereas the cladding is an austenitic stainless steel (18 Cr-8 Ni).

A chemical element of special interest in both the vessel base material and associated welds is copper, since it enhances radiation damage, which results in reduced fracture toughness. High concentrations of copper exist in some welds because the weld wire was plated with copper; this element is also present as an impurity in the base material.

The chemical composition of the vessel materials influences all the parameters in Table 7.1, while the various vessel heat treatments (tempering and stress relieving) affect primarily σ_y, σ_u, and RTNDT.

7.2.2 Temperature Dependence of Properties

The temperature dependences of the parameters in Table 7.1 are reasonably well known, and each parameter (with the exception of RTNDT and ν, which has a negligible temperature dependence) or an appropriate combination thereof, is included in the ASME Pressure Vessel Code as a function of temperature. Values for k, $k/\rho c_p$, α, E, σ_y, and σ_u

Table 7.2 Materials Used in the Fabrication of U.S. Commercial Reactor Vessels

	Wgt. Percent Composition for Designated Materials		
Element	Plate SA 302 GR B	Plate SA 533 GR B, Cl 1	Forging SA 508 Cl 1
C	0.25 max	0.25 max	0.27 max
Cr	—	—	0.25-0.45
Ni	—	0.40-0.70	0.50-1.00
Mo	0.45-0.60	0.45-0.60	0.55-0.70
Mn	1.15-1.50	1.15-1.50	0.50-0.90
Si	0.15-0.30	0.15-0.30	0.15-0.35
P	0.035 max	0.035 max	0.025 max
S	0.040 max	0.040 max	0.025 max
Fe	balance	balance	balance

as functions of temperature are included in ASME Section III, while K_{Ic} and K_{Ia} as functions of T - RTNDT are included in ASME Section XI for temperatures (T) less than that associated with the upper shelf[1,2]. The uncertainty in the parameters included in ASME Section III is not large and is expected to have a negligible effect on the evaluation of vessel integrity. However, the uncertainties in K_{Ic} and K_{Ia} vs T - RTNDT is substantial, and there is also a significant uncertainty (approximately ±20°F) in the determination of RTNDT.

The curves for K_{Ic} and K_{Ia} vs T - RTNDT in ASME Section XI represent the lower bound of a limited amount of data that were obtained some years ago for A533 and A508 material. The use of this lower bound would appear to be conservative; however, recent experiments with large test cylinders[3,4] indicate that long cracks in large structures will usually behave in accordance with the lower bound of data obtained from a large number of small (1T to 3T-CT) specimens. The uncertainty in the ASME Code lower bound is under investigation at this time.

The uncertainty associated with the use of RTNDT as a normalization factor is also under investigation. An alternative to relying on RTNDT is to determine, through laboratory testing, K_{Ic} and K_{Ia} vs T for each vessel. However, this approach appears to be impractical because of the large number of specimens required, since RTNDT is a function of fluence.

During a reactor vessel thermal transient of medium duration, the outer portion of the vessel wall remains at temperatures corresponding to ductile behaviour (i.e., the upper-shelf portion of the K_{Ic} vs T curve). It is not likely that cleavage (brittle) fracture can proceed through this zone; however, the crack may tear at relatively low crack-tip velocities to a depth at which plastic instability is achieved. Tearing-resistance material property data are required for an accurate analysis, and such data are, at present, very limited. An alternative approach currently used is to assume what appears to be a conservative upper-shelf toughness that is essentially independent of temperature and fluence; this upper shelf value is then compared to the calculated stress intensity factor. For very severe accidents such an approach is probably adequate, but for less severe cases the tearing resistance may terminate crack propagation. The degree of conservatism in the present model for the less severe cases is not known.

7.2.3 Dependence of Material Properties on Fast-Neutron Fluence

Of the material properties listed in Table 7.1, those that have a significant dependence on fast-neutron fluence are σ_y, σ_u, K_{Ic}, K_{Ia}, and RTNDT. Radiation damage increases σ_y and, to a lesser extent, σ_u, while K_{Ic} and K_{Ia} are decreased and RTNDT is increased. The average of σ_y and σ_u is used at the conclusion of the fracture-mechanics analysis to see if the uncracked ligament has become plastic under pressure loading. Usually, strength values for the unirradiated material are used, and this approach introduces some conservatism. Some data on elevated strength are available, and their use in the analysis would result in a higher permissible pressure during a thermal transient.

In accordance with ASME code procedure, the decreases in K_{Ic} and K_{Ia} due to radiation damage are estimated by shifting the K_{Ic} and K_{Ia} vs T curves along the temperature axis by an amount $\Delta RTNDT = f(F, Cu, P)$, where F = fast-neutron fluence ($E \geq 1$ MeV) and Cu and P are the copper and phosphorous concentrations, respectively. Values for $\Delta RTNDT = f(F, Cu, P)$ are included in Reg. Guide 1.99, Rev. 1,[5] which was thought to be conservative at the time of its issuance. A more recent evaluation[6] of the available data indicates that the ΔRTNDT for materials containing a high concentration of nickel, which appears to enhance the effect of copper on radiation damage, agrees rather well with Reg. Guide 1.99, while lower concentrations of nickel result in conservative values for the reference temperature shift. There are indications that many of the welds in the older PWR reactor vessels have high concentrations of nickel, and thus estimates of ΔRTNDT from Reg. Guide 1.99 presumably are not excessively conservative. However, the data base is much smaller than would be desired and will take some time to increase substantially, even though surveillance specimens from power reactors are becoming available and irradiation programs at materials testing reactors are under way.

To date, radiation damage to the cladding is of little concern to the analysis of overcooling accidents because it has been assumed that the initial flaw will extend through the vessel cladding into the base material, and also that the flaw will be very long on the surface so that cladding resistance to crack extension is not important. However, an assumption of long initial flaws may be unnecessarily conservative, since the presence of cladding may prevent short flaws from extending, particularly if the cladding retains its high tearing resistance at high fluences. There is a limited amount of data[7] for weld cladding that indicates a substantial reduction in Charpy upper-shelf energy (~100 to 30 ft-lb) at a fluence of $\sim 8 \times 10^{19}$ n/cm^2 and an irradiation temperature of 550°F.[8] Thus, it is not clear that the cladding will prevent short flaws from growing long.

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8. It is of interest to note that because of a recent change in core loading, the anticipated fluence by the end of 32 EFPY for Oconee-1 is ~1.1×10^{19} neutrons/cm^2 (personal communication, Duke Power Company, October 1981).

INTEGRITY OF REACTOR VESSELS DURING OVERCOOLING AC'IDENTS

8.1 Description of Basic Problems

During an overcooling transient in a PWR the reactor pressure vessel is subjected to a thermal shock in the sense that thermal stresses are created in the vessel wall as a result of rapid removal of heat from its inner surface. The thermal stresses are superimposed on the pressure stresses, with a result that the net stresses are positive (tensile) at and near the inner surface of the wall and are substantially lower and perhaps negative elsewhere, depending on the magnitude of the pressure stress. The concern over the high tensile stresses near the inner surface is that they result in high stress intensity factors (K_I) for any inner-surface flaws which may be present. To compound the matter, the reduced temperature and the relatively high fast-neutron fluence near the inner surface result in relatively low fracture toughness values (K_{Ic} and K_{Ia}) for the vessel material in the same area. Thus, there is a possibility of crack propagation. The positive gradient in temperature, combined with the negative gradients in stress and fluence through the wall, tends to provide a mechanism for crack arrest deeper in the wall. However, if the crack is very long on the surface and propagates deep enough, the remaining vessel ligament will become plastic and the vessel internal pressure will ultimately result in rupture of the vessel. Thus, for each thermal transient there will be a maximum permissible pressure that is a function of time.

Crack propagation may also be limited by a phenomenon referred to as warm prestressing (WPS), which has been demonstrated in the laboratory with small specimens[1] and also in a rather large, thick-walled cylinder during a thermal shock experiment.[2] In such cases, WPS simply refers to the inability of a crack to initiate while K_I is decreasing with time, i.e., while the crack is closing. While this special situation is encountered during some specific overcooling accidents, caution must be exercised in taking credit for WPS because changes in the pressure that affect little else can delay or eliminate the requisite conditions for WPS.

The integrity of a reactor vessel during a postulated overcooling event is evaluated in terms of the continued ability of the vessel to contain the coolant in such a way that melting of the reactor fuel will not occur. Generally speaking, this means that the water level must be maintained above the core, and to do this there must not be a significant breach in the vessel wall below the level established by the top of the core. Therefore, it is necessary to determine if a preexistent flaw will propagate through the wall, and if it will, to estimate the probable size of the breach and its resistance to leakage.

The investigative effort thus far has been directed at understanding the behavior of flaws during thermal, pressure, and thermal-plus-pressure loading conditions. It has been assumed on the basis of limited available data that if the temperature of a major portion of the coolant in the primary system is well above 212°F at the time a long flaw penetrates the wall, the final opening may be excessive in the sense that flooding of the core could not be maintained. Methods for estimating the size of the opening more accurately will be evaluated in the near future.

In the following paragraphs a calculational model for predicting crack behavior during overcooling accidents is described, and a summary of results for specific accidents is presented and discussed. The reactor plant analyzed for these studies is Oconee-1, and the postulated accidents include a main steam line break, a turbine trip followed by stuck-open bypass valves, a small-break LOCA, a Rancho Seco-type transient, and a large-break LOCA.

8.2 Calculational Model

The calculational model consists of three basic parts: a thermal analysis of the vessel wall, a stress analysis, and a fracture-mechanics analysis. The thermal analysis is performed for cylindrical geometry, is one-dimensional (radial direction), includes an insulated outer surface, and accepts a transient coolant temperature at the wall's inner surface. A time-independent inner-surface thermal resistance is used that is the sum of the fluid-film resistance and the cladding resistance, in which case the heat capacity of the cladding is ignored.

Since an important input to the thermal analysis is the temperature of the coolant in the downcomer region, some of the assumptions used in obtaining this temperature need to be mentioned. Depending on the nature of the overcooling accident, the temperatures of the coolant entering the downcomer may be different for the different inlet coolant pipes. Thus, there can be azimuthal and axial variations in the downcomer coolant temperature. An accurate determination of the temperature distribution as a function of time would be very difficult, and the subsequent use of two- or three-dimensional stress and fracture-mechanics computations would be impractical for a parametric-type analysis. An additional complication in this regard for B&W reactors is that relatively warm water from the core outlet may enter the upper portion of the downcomer region through the vent valves and may thus raise the temperature of the downcomer coolant. For the purpose of the present analysis the coolant temperature (temperature transient) used as input to the vessel thermal analysis corresponds, with one exception, to the lowest of the cold legs (inlet lines). The degree of conservatism associated with this assumption is unknown.

The stress analysis is also one-dimensional and is performed for a cylinder; the cladding is excluded, as is the flaw, consistent with the method used for calculating the stress intensity factor. Loads on the cylinder consist of a radial temperature distribution and internal pressure, both of which are treated as functions of time.

Linear elastic fracture mechanics (LEFM) is used for predicting flaw behavior. The flaw assumed for this particular study is an inner-surface, long, axially-oriented, sharp crack of uniform depth along its length. Thus, an accurate, two-dimensional (radial and azimuthal) model can be used. Consideration of a long axial surface flaw is realistic and necessary in the sense that short flaws will tend to become long flaws under thermal-shock loading, and the stress intensity factor is greater for long axial surface flaws than for any others. The two-dimensional model does, however, introduce some conservatism since there exists an axial gradient in fluence (and hence in toughness) that is ignored but which will provide some additional resistance to crack propagation. Furthermore, the cladding, which tends to be a much tougher material than the base material, may suppress the surface extension of short flaws that has been predicted and observed in the absence of cladding.

As already mentioned, although the cladding is included in the thermal analysis it is not included in the fracture-mechanics analysis. The presence of cladding reduces slightly the tensile stress in the base material during a thermal transient. However, if the flaw extends through the cladding, the K_I value is significantly greater than if the cladding were ignored, particularly for shallow flaws. Thus, the minimum critical crack depth for crack initiation would be less and the threshold fluence for crack initiation and vessel failure would also be less. A detailed quantitative assessment of this cladding effect is not yet available.

Fracture-toughness curves (K_{Ic} and K_{Ia} vs T - RTNDT) for this study were taken from Sect. XI of the ASME Code,[3] and an upper-shelf toughness of 200 ksi$\sqrt{\text{in.}}$ was added for both K_{Ic} and K_{Ia}.
Input to the fracture-mechanics analysis includes (1) the temperature and fast-neutron fluence distributions through the wall, (2) the thermal and pressure stresses without the presence of the flaw, and (3) the copper (Cu) and phosphorous (P) concentrations. The temperature and fluence distributions, coupled with the Cu and P concentrations, are used to calculate K_{Ic} and K_{Ia} radial distributions at various times in the transient, and the stresses are used to calculate K_I values for a number of crack depths, ranging from ~3 to 90% of the wall thickness.

The thermal, stress, and fracture-mechanics analyses were performed with the OCA-I computer code,[4] which uses a superposition technique to accurately calculate K_I values for long flaws using stresses for the unflawed cylinder. The purpose in using the superposition technique was to achieve the accuracy of a finite-element analysis at a fraction of the usual cost, thus making parametric studies feasible. A block diagram describing the code is shown in Fig. 8.1. Required input for the OCA-I analysis is also indicated in Fig. 8.1, and specific values used for the Oconee-1 studies included herein are given in Table 8.1.

For these preliminary studies only five transients were analyzed, but for each transient several inner-surface-fluence values were included so that the threshold fluence (and thus the number of years of operation) for incipient crack initiation could be estimated.

8.3 Transients Considered for Oconee-1

8.3.1 Main Steam Line Break

Time-dependent pressure and temperature curves for this case were submitted to ORNL by Brookhaven National Laboratory (BNL) on August 14, 1981[5] and are given in Fig. 8.2 and Table 8.2 (corresponds to case 3 in Section 5). For the thermal analysis of the vessel the fluid-film heat-transfer coefficient was assumed to be 1000 Btu/hr·ft^2·°F, which corresponds to full-flow conditions (primary system) and a total surface conductance of 330 Btu/hr·ft^2·°F.

8.3.2 Rancho Seco Transient

The March 1978 Rancho Seco transient was made available to ORNL by NRC in January 1981[6] and is shown graphically in Fig. 8.3 and in tabular form in Table 8.3. As described in Sect. 5, the coolant temperature is measured upstream of the injection point for the HPIS and is thus probably somewhat higher than actually exists at the entrance to the downcomer. The fluid-film heat-transfer coefficient at the vessel wall was assumed to be 1000 Btu/hr·ft^2·°F.

8.3.3 Turbine Trip Followed by Stuck-Open Bypass Valves

This transient, which corresponds to case [illegible] in Section 5, was submitted to NRC by BNL in July 1980[7] and to ORNL by BNL in August 1980[8] and is shown graphically in Fig. 8.4 and in more detail in Table 8.4. The portion of the transient beyond 1240 s was added by ORNL, assuming that the HPIS would pump against the relief valve setting (2500 psi) and that the temperature of the coolant in the downcomer would remain at 140°F. The fluid-film heat-transfer coefficient was again assumed to be 1000 Btu/hr·ft^2·°F.

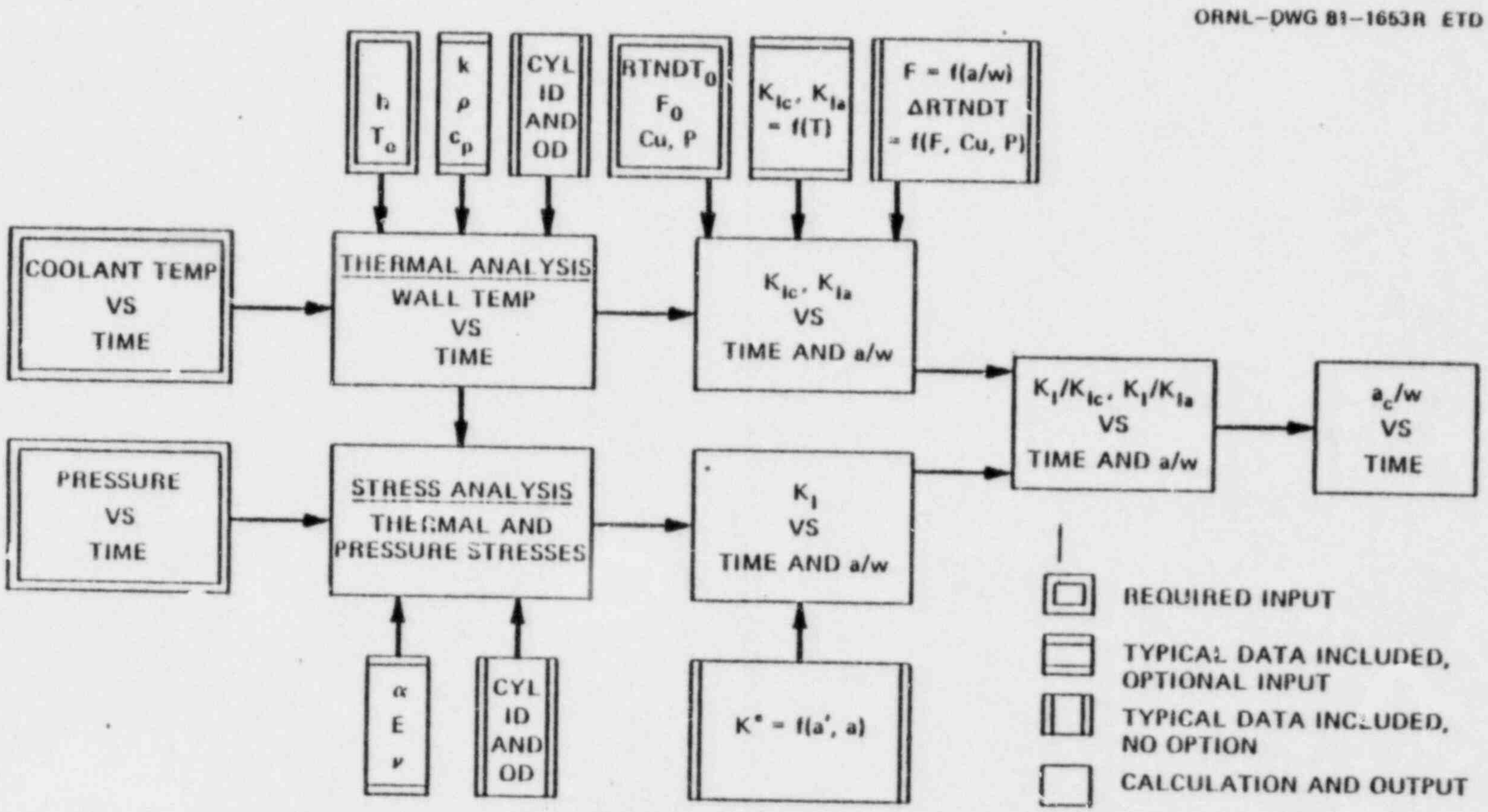

Fig. 8-1. Block Diagram of the OCA-I Computer Code, Indicating Basic Input Calculations, and Output.

Table 8.1. Input to OCA-I for Oconee-1 Analysis

Vessel Dimensions, in.	
Outside Dia.	189
Inside Dia.	172
Coolant Temp vs Time	Specific Accident
Pressure vs Time	Specific Accident
Heat Transfer Coeff. (h), $Btu/hr \cdot ft^2 \cdot {}^\circ F$	
Large-break LOCA	200
Others	330
Initial Wall Temp, °F	550
$RTNDT_o^a$, °F	40
Copper Concentration (Cu), %	0.31
Phosphorous Concentration (P), %	0.012
K_{Ic} and K_{Ia} Upper Shelf, ksi $\sqrt{in.}$	200
Fluence at Inner Surface (F_o)	Range of Values

[a] $RTNDT_o$ = zero-fluence RTNDT (initial value)

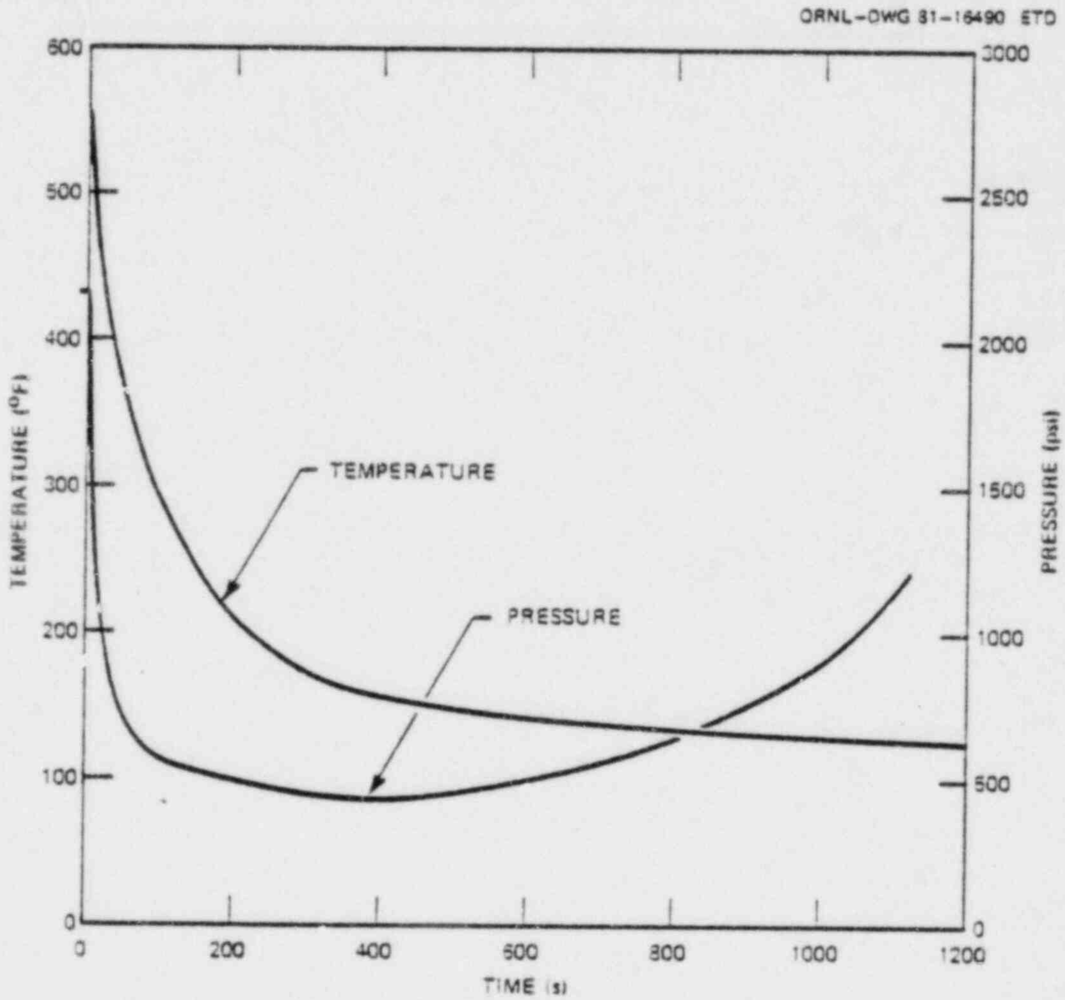

Fig. 8-2. Main Steam line break temperature and pressure transients (HPIS remains active).

Table 8.2. Main Steam Line Break Temperature and Pressure Transients (HPIS Remains Activated)

Time (s)	PZR Level (ft)	Primary Pressure (psia)
0	18.23	2157.8
1	17.44	2139.1
5	11.01	1964.35
10	2.10	1677.02
15	0.0	1440.25
20	0.0	1323.79
25	0.0	1216.75
30	0.0	1106.18
35	0.0	904.86
40	0.0	760.18
45	0.0	743.5
50	0.0	725.19
60	0.0	689.04
70	0.0	650.19
80	0.0	610.61
90	0.0	578.52
100	0.0	559.57
120	0.0	530.47
140	0.0	513.61
160	0.0	523.63
180	0.0	517.91
200	0.0	507.52
220	0.0	497.72
240	0.0	487.96
260	0.0	478.35
[illegible]0	0.0	468.92
[illegible]0	0.0	459.61
350	0.0	438.92
400	0.22	429.69
450	1.40	443.36
500	2.69	459.39
550	4.12	478.53
600	5.67	501.16
650	7.30	527.56
700	9.04	558.72
750	10.84	595.07
800	12.69	637.46
850	14.56	686.83
900	16.43	747.1
950	18.30	818.56
1000	20.14	903.93
1050	21.96	1006.48
1103.58	23.84	1139.87
1123.18	24.51	1195.84

ORNL-DWG 81-8080 ETD

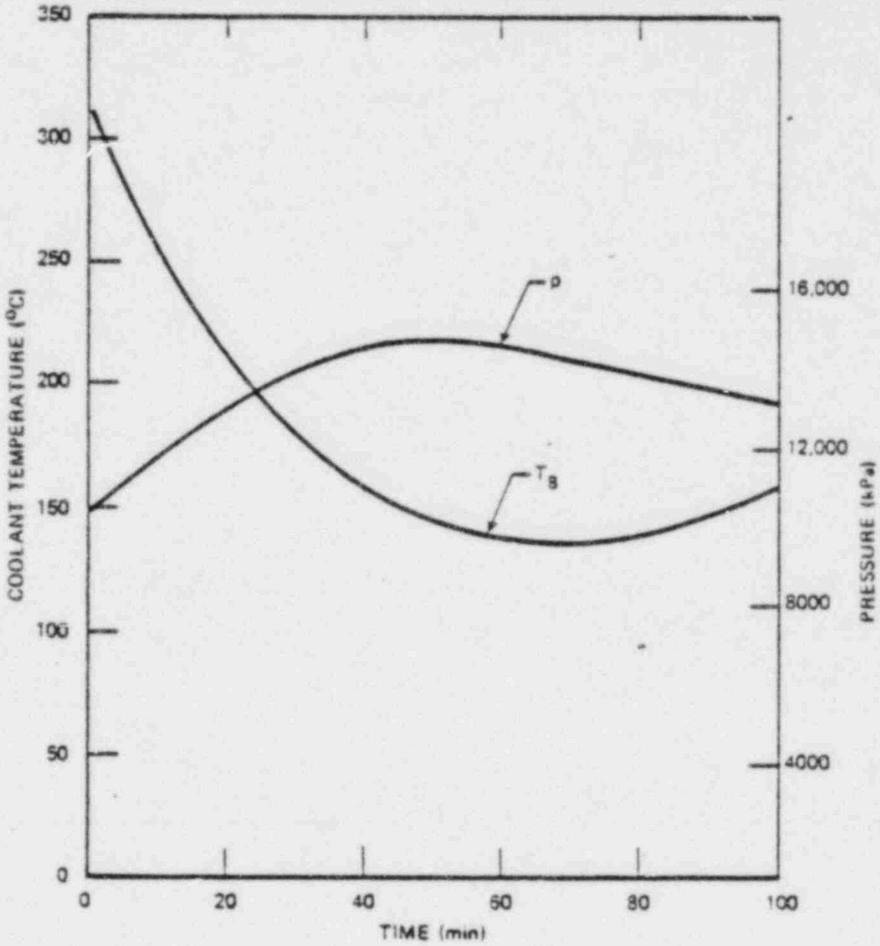

Fig. 8.3. Temperature and pressure transients for Rancho Seco.

Table 8.3. Rancho Seco Temperature and Pressure Transients

Time (min)	Temperature (°F)	Pressure (psi)
0	590	1500
10	490	1710
20	412	1880
30	356	2020
40	318	2110
50	296	2130
60	282	2100
70	280	2050
80	284	2000
90	299	1950
100	320	1900

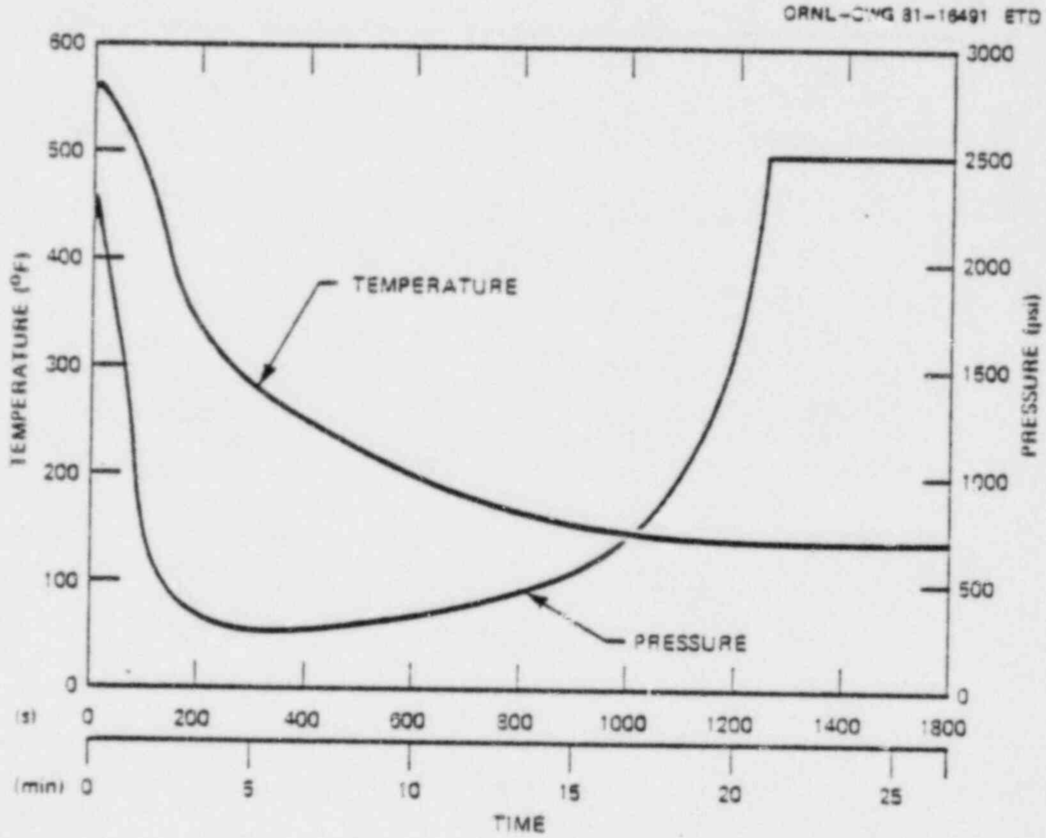

Fig. 8-4. Turbine trip with stuck-open bypass values (Scram Worth = 0.061 $\Delta k/k$): Temperature and Pressure Transients.

Table 8.4. Turbine Trip with Stuck-Open Bypass Valves (Scram Worth = 0.061 $\Delta k/k$): Temperature and Pressure Transients

Time (s)	Core Inlet Temperature (°F)	Primary System Pressure (psia)
0	556.00	2,192.00
5	563.64	2,289.31
55	528.73	1,570.92
125	444.09	523.95
215	328.42	320.58
340	271.34	275.21
630	197.45	365.93
800	167.70	463.54
900	157.08	561.81
1000	149.97	723.80
1050	147.34	847.90
1100	145.20	1,020.84
1150	143.45	1,273.30
1200	142.06	1,663.70
1240	141.17	2,149.54

8.3.4 Small-Break LOCA

The SBLOCA case was defined and the thermal-hydraulic analysis was performed by Los Alamos National Laboratory; it was reviewed by NRC in June 1981[9] (case 5 in Section 5). The break was assumed to be in the cold leg downstream of the main circulating pump and ahead of the HPIS injection nozzle and was sized at 10% of the pipe area. The thermal-hydraulic analysis was performed for the first 250 s only, and the resulting pressure and temperature transients are shown in Fig. 8.5 and Table 8.5. The temperature given is that calculated for the top of the downcomer; since the main circulating pumps were assumed to be tripped at 15 s into the transient and the HPIS was assumed to inject coolant for the duration of the transient, this coolant temperature is probably higher than would actually exist locally at the vessel wall, i.e., channeling of the HPIS coolant would occur.

8.3.5 Large-Break LOCA

The LBLOCA has been under detailed investigation for several years[10] and differs from the other transients considered in that no repressurizaton of the primary system takes place, and the downcomer temperature transient consists of an essentially step change in temperature from normal operating temperature (550°F) to ~70°F. The fluid-film heat-transfer coefficient used in the thermal analysis of the vessel corresponds to free convection and was estimated to be 300 Btu/hr·ft^2·°F, which corresponds to a total surface conductance of ~200 Btu/hr·ft^2·°F.

8.4 Results of Fracture-Mechanics Analyses

The fracture-mechanics analysis will indicate one of three possible results for each specific case: (1) there will be no crack initiation for any reasonable assumed preexistent crack depth; (2) crack initiation will occur, but the crack will arrest permanently; or (3) crack initiation will occur and the crack will penetrate the wall.

The results of the analysis are quite sensitive to the radiation-induced reduction in toughness and thus to the fast-neutron fluence, which is a function of the operating time of the reactor. Therefore, the results are summarized in terms of the threshold fluence for incipient crack initiation and for failure of the vessel, and this is done for two cases: (1) assuming WPS to be effective (if appropriate conditions exist), and (2) assuming WPS not to be effective (even if appropriate conditions do exist). 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.

The summary of results presented in Table 8.6 shows that for all cases analyzed the minimum critical crack depths for initiation are in the

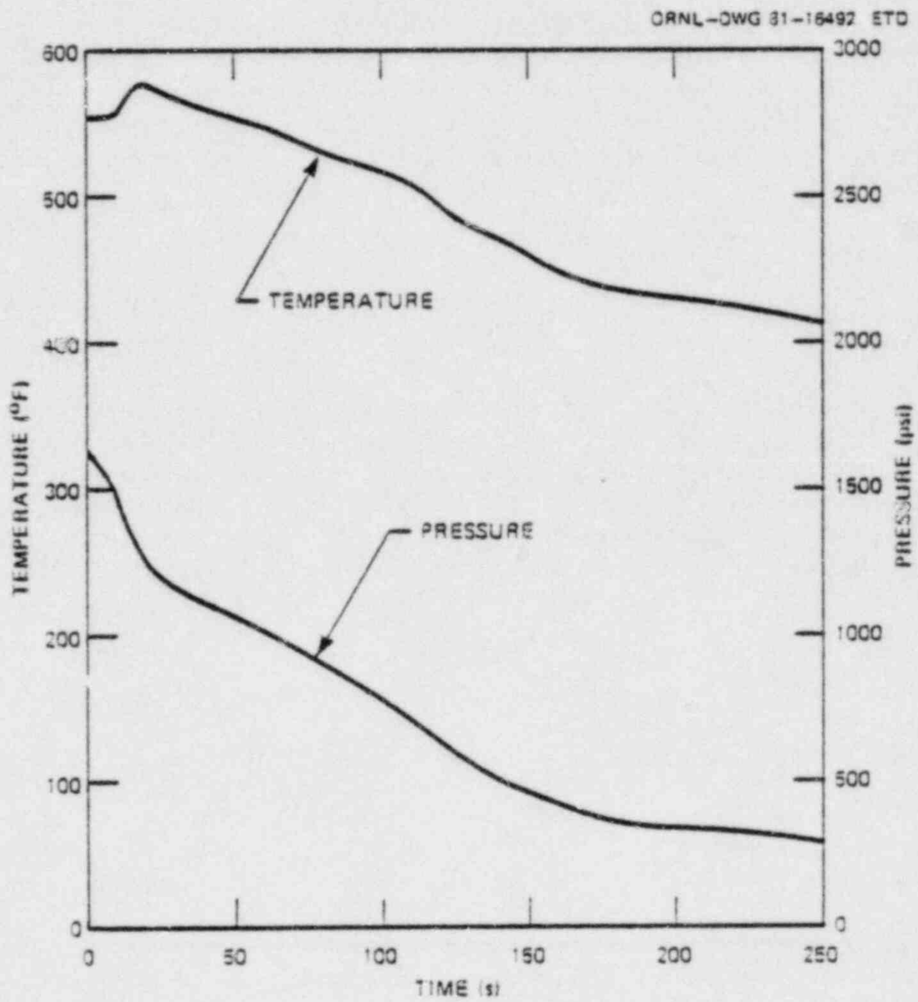

Fig. 8-5. Small-Break LOCA temperature and pressure transients.

Table 8.5. Small-Break LOCA Temperature and Pressure Transients

Time (s)	P (ksi)	T (°F)
0	1.63	553
12	1.45	558
17.3	--	576
25	1.20	570
50	1.06	553
75	0.930	537
100	0.777	516
125	0.590	486
150	0.457	459
175	0.383	439
200	0.348	430
225	0.322	423
250	0.290	415

range of 0.17-1.3 in. This implies that at least some of the flaws, because of their size (upper end of the range), might have a high probability of being detected by nondestructive means. However, for fluences somewhat greater than those associated with incipient initiation and failure the upper and of the range is much lower. The calculated critical crack depth would be further reduced relative to the values in Table 8.6 by including the effect of cladding in the fracture mechanics analysis, assuming, as we are, that the crack extends through the cladding. As mentioned earlier, 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.

In evaluating the data in Tables 8.6 and 8.7 there is a distinction that must be made between the large-break LOCA, and the other cases. Since the LBLOCA does not involve repressurization, a long, axially oriented flaw presumably would not extend completely through the wall, and even if it did the crack would remain tight and thus leakage of coolant presumably would be negligible. For the other cases the primary system pressure is high enough, consistent with assumptions made, to force the crack all the way through the wall. Furthermore, since the system is pressurized, the temperature of the coolant could be high enough to result in sufficient energy release during blowdown to open the crack substantially. As mentioned earlier, a detailed analysis of crack opening under these circumstances has not yet been performed.

As indicated in the tables, WPS was predicted for each of the cases considered. It may be reasonable to take advantage of WPS for the LBLOCA since, by definition, there is no repressurization; however, for the other cases, variations in the repressurization could preclude conditions for WPS without making significant differences in the results otherwise. Thus, one cannot necessarily depend on WPS to reduce the consequences of the transient.

The fluences listed in Table 8.6 correspond to those at the inner surface of the vessel wall at locations having the specified copper concentrations. Thus, to determine the number of EFPYs in Table 8.7 it is necessary to know the fluence rate (fluence per EFPY) at the same locations. To establish the most limiting location one must consider the combined effect of fluence, copper concentration and initial RTNDT ($RTNDT_o$). The location that would tend to have the highest RTNDT ($RTNDT_o + \Delta RTNDT$) would be the likely choice. Such a location was established for Oconee-1 in Ref. 11, and the corresponding fluence rate is 0.046×10^{19} neutrons/cm^2/EFPY. According to the information in Sect. 6 the uncertainty in this value is ±50%. The mean value was used to obtain the threshold times (EFPYs) to vessel failure listed in Table 8.7.

Table 8.6. Summary of Flaw Behavior Characteristics for Several Hypothetical Oconee-I Overcooling Accidents (refer to the list of nomenclature for this table on the following page)

	F_{ii} ($n/cm^2 \times 10^{19}$)	$(a/w)_{ii}^{a}$	t_{ii} (min)	$(a/w)_{aii}$	F_{if} ($n/cm^2 \times 10^{19}$)	$(a/w)_{if}^{a}$	t_{if} (min)
				Large-Break LOCA[b]			
WPS	0.4	∿0.04	5.5	0.15	0.9	0.02-0.22	1.5-8
Without WPS	0.15	0.1	30	0.5	0.2	0.04-0.18	14-45
				Rancho Seco			
WPS	1.5	0.1	40	1.0	1.5	0.1	40
Without WPS	0.9	0.1	65	1.0	0.9	0.1	65
				Turbine Trip/Open Bypass Valves			
WPS	0.2[c]	0.06	22	1.0	0.2[c]	0.06	22
Without WPS	0.13[d]	0.15	60	1.0	0.13[d]	0.15	60
				Main Steamline Break			
WPS	0.4	0.04	9	1.0	0.4	0.04	9
Without WPS	0.2[d]	0.06	18	1.0	0.2[d]	0.06	18
				Small-Break LOCA			
	No initiation[e]						

[a]If a range of crack sizes is not shown, shallower and deeper flaws will initiate at higher fluences.

[b]For this case, failure refers to crack penetration beyond a/w = 0.9. Presumably the crack will not actually penetrate the outer surface.

[c]Crack depths greater than a/w = 0.2 ignored.

[d]If the transient time were extended beyond 60 min., F_i would be less.

[e]Duration of thermal-hydraulic transient simulated too short to permit meaningful fracture-mechanics analysis.

Nomenclature for Table 8.6

$(a/w)_{ii}$	fractional crack depth for incipient initiation
$(a/w)_{aii}$	fractional crack depth for final arrest following incipient initiation
$(a/w)_{if}$	fractional crack depth for first initiation that results in vessel failure (corresponds to threshold fluence for failure)
F_{ii}	threshold fluence at inner surface of vessel wall for incipient crack initiation
F_{if}	threshold fluence at inner surface of vessel wall for incipient vessel failure
t_{ii}	time in transient for incipient crack initiation
t_{if}	time in transient for incipient vessel failure

Table 8.7. Estimated Threshold Times for Vessel Failure for a B&W-type Reactor for Various Overcooling Events

Postulated Transient	Threshold Time[a] (EFPY[b])	Comments and Qualifications
Large-Break LOCA	· 20	WPS assumed effective (4 EFPY w/o WPS). Crack not expected to penetrate RV wall; vessel integrity expected to be maintained, since repressurization does not occur.
Rancho Seco	20	WPS not assumed effective (33 EFPY if it were). Through-wall crack predicted.
Turbine Trip with Open Bypass Valves (RFT)[c]	3	WPS not assumed effective (4 EFPY if it were). Through-wall crack predicted.
Main Steam Line Break[c]	4	WPS not assumed effective (8 EFPY if it were). Through-wall crack predicted.
Small-Break LOCA	–	Thermal-hydraulic forcing functions calculated to only 250 s (at which time temperature and pressure are still decreasing), thereby preventing meaningful analysis of crack propagation in RV.

[a]Effective full-power years at which failure of the vessel is predicted, given the pressure and thermal driving functions presently predicted for the transients, and the assumptions used in this study.

[b]Based on a fluence accumulation rate value[11] of 0.046×10^{19} n/cm^2/EFPY. Depending on the specifics of surveillance programs and fuel management schemes, this value may have an associated uncertainty of as much as ±50%.

[c]Failure predictions based on thermal-hydraulic calculations containing conservative assumptions.

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1. F. J. Loss, R. A. Gray, Jr., and J. R. Hawthorne, Significance of Warm Prestress to Crack Initiation During Thermal Shock, NRL/NUREG Report 8165 (Sept. 29, 1977).

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5. Letter to R. Kryter, ORNL, from R. J. Cerbone, BNL, Aug. 14, 1981.

6. Letter to R. D. Cheverton, ORNL, from J. Strosnider, NRC, January 30, 1981.

7. Letter to Novak Zuber, NRC, from M. M. Levine, BNL, July 1, 1980.

8. Letter to R. D. Cheverton, ORNL, from M. M. Levine, BNL, August 11, 1980.

9. Letter to C. Z. Serpan, NRC, from S. Fabic, NRC, June, 22, 1981.

10. R. D. Cheverton, S. K. Iskander and S. E. Bolt, Applicability of LEFM to the Analysis of PWR Vessels Under LOCA-ECC Thermal-Shock Conditions, ORNL/NUREG-40 (October 1978).

11. J. Strosnider, NRC, personal communication (September 1981).

9.0 POSSIBLE MITIGATIVE MEASURES

9.1 Operator Actions

In the case of the Oconee-1 control system, reactor operators clearly have a manual capability to terminate excessive feedwater for a wide variety of failures. However, such actions require that the operator correctly recognize overcooling problems at a time early enough in the transient that his resulting response is effective. Diverse sources of information are available to the operators, from which a determination of overcooling can be made. It is therefore possible that operator response can be an effective deterrent to this problem.

It is conceivable that the operator might be able to control the outlet subcooling for certain accidents, as B&W proposes. How practical this is, considering instrument locations and fluid transport times (particularly if the reactor coolant pumps are tripped), needs to be evaluated.

The questions of whether and/or when to trip the reactor coolant pumps in overcooling upsets need evaluation. Tripping the pumps will raise the temperature and delay the influx of cold water to the vessel for steam-generator-driven transients; however, maintaining a properly cooled core and promoting good downcomer mixing may necessitate leaving the pumps on. It is evident that the operator needs a definitive indicator of adequate core cooling and vessel wall temperatures to achieve a proper balance between concerns for core cooling and vessel overcoolings.

9.2 System Changes

9.2.1 Oconee Changes

Our review of the Oconee-1 control system revealed that several of the upgrades already performed will act to reduce the likelihood and extent of excessive feedwater transients in this plant. Among these changes are upgrades to the instrument and control system power supplies, identification of diverse information channels for operator use during power supply failures, automatic feedwater pump trip on loss of ICS power, and steam generator high level limit trips for the main feedwater pumps. This reduces the probability of excessive feedwater being supplied to that of two failures of the control system and failure of the operator to take corrective action or overt operator error in manually actuating the feedwater.

Other potential changes can be identifed. An analysis of the condensate booster pump response following main feedwater pump trip could be used to determine whether or not other feedwater system control actions should be initiated on loss of ICS power. Similarly, analysis could be performed to determine whether a main feedwater trip based on a specified pressure-temperature envelope could be used to prevent inadvertent overcooling. A more detailed systems analysis could possibly identify other alternatives.

9.2.2 General Changes

There are several other changes to the reactor system which deserve evaluation for their effectiveness, cost, and practicality of implementation. These are outlined below.

Borated Water Storage Tank (BWST) Temperature. Increasing the temperature of the BWST fluid would reduce the degree of overcooling caused by actuation of the safety injection systems. However, this measure obviously has limited effectiveness, since some tanks probably cannot be heated above ~80°F and, in any case, maintenance of temperatures above ~200°F would be impossible without tank pressurization.

Feedwater Train Storage. Limiting the amount of water available to the feedwater system would obviously reduce the severity of the steam-generator-driven transients. However, there is a trade-off here with practical requirements for normal plant operation.

Containment Flooding. This "fix" has been proposed and discussed in connection with other major accidents. It would obviously mitigate the consequences of a reactor vessel breach; however, inadvertent actuation of such a system might itself produce a vessel breach through rapid and extreme overcooling of the RV outer wall.

9.3 Restoring Pressure Vessel Fracture Toughness By Annealing

From a vessel design point of view, the most desirable solution to the overcooling-accident vessel-integrity problem is to restore the vessel material fracture toughness, which is gradually reduced during reactor operation as a result of exposure of a vessel wall to fast neutrons. Studies that have been underway for several years indicate that the toughness can be restored by annealing at temperatures in the range of 750-850°F for a period of approximately 200 hours[1,2]. The results of studies conducted by Westinghouse, under contract to EPRI, indicate that conducting the annealing treatment of the irradiated vessel is practical for some and perhaps most of the vessels in service today[1].

REFERENCES - CHAPTER 9

[1] T. R. Mager (Westinghouse), personal communication to R. D. Cheverton (ORNL), September 29, 1981.

[2] F. Loss et al. (NRL).

10.0 CONCLUDING REMARKS AND RECOMMENDATIONS FOR FURTHER WORK

10.1 Concluding Remarks

Despite a fair degree of recent effort in the study of pressurized thermal shock phenomena by a number of knowledgeable groups, the true severity of the threat is, at present, very difficult to ascertain with confidence. The principal problems contributing to uncertainty are:

- The computer codes presently being used to simulate the hypothetical overcooling transients were not designed to treat some of the phenomena that take place and hence produce inaccurate (sometimes nonphysical) thermal-hydraulic forcing functions under certain circumstances. The results of the fracture-mechanics analysis used to predict vessel failure are known to be sensitive to the temporal behavior of these forcing functions.

- The temperature indications in the lone set of actual plant data available (Rancho Seco) provide only a nominal indication of true RV wall conditions (they could be too high or too low by an indeterminate amount), and the chart-recorded pressure traces are made suspect by the presence of large "spikes" of unknown and possibly nonphysical origin.

- The thermal/stress/fracture-mechanics analyses presently used to predict crack propagation resulting from the temperature and pressure forcing functions have limitations (e.g., 1-D thermal and stress analysis; lack of treatment of azimuthal and axial variations in downcomer coolant temperature; inability to account for the axial gradient in wall fluence; lack of treatment of vessel cladding in fracture-mechanics analysis) which introduce uncertainty of an unknown magnitude in the results.

- The fluence at the vessel wall and at critical welds is probably known only to an accuracy of ±30% (perhaps ±50%), and this implies an uncertainty of like magnitude in the vessel "life remaining" figures.

- The probabilities of occurrence for various overcooling accident initiating events have associated uncertainties of at least plus-or-minus one order of magnitude, and the conditional probabilities for correct subsequent operator diagnosis of a transient, timeliness and correctness of operator response, appropriate automated control and safety features responses, and the like are at present undetermined.

Nonetheless, for all their shortcomings, the analyses at hand are the best presently available on a nonproprietary basis, and, owing to the apparent severity of the outcomes predicted from the limited number of overcooling scenarios studied, it is our opinion that pressurized thermal shock must be regarded as a serious potential threat and merits a great deal more study using refined techniques.

10.2 Recommendations for Further Work

In order to reduce the magnitudes of the uncertainties described above, we recommend that additional work be undertaken in the following areas:

- Refinement of thermal-hydraulic simulation codes and associated models (in particular, treatment of the feed train, fluid mixing, the control system, primary coolant system repressurization and flow distribution, two-phase phenomena, and the heat capacity of heavy primary metal).

- Refinement of vessel thermal/stress/fracture-mechanics analysis techniques (in particular, a consideration of higher dimensionality in several of the variables treated, and inclusion of the vessel cladding in the fracture mechanics).

- Refinement of the analytical methods and surveillance capsule data assessment procedures required to estimate fast-neutron fluence in selected areas of the RV wall, in order that state-of-the-art accuracies ($\pm$10%) may be realized.

- A thorough study of the probability structure of the various intertwined occurrences (among them, normal plant maneuvers, chance events, equipment and operator failures, plant recovery actions, etc.) that are necessary to produce the severe thermal shock conditions that constitute a serious threat to RV integrity.

Some facets of this recommended work are known to be in progress by the NRC and reactor owner's groups or will be initiated in FY 1982.

APPENDIX A
CAPSULE DESCRIPTION OF DATA BASES EXAMINED

1. BHRA - Fluid Engineering

BHRA Fluid Engineering provides indexing and abstracting of world-wide information on all aspects of fluid engineering, including statics and dynamics, and laminar and turbulent flow. Theoretical research is covered, as well as the latest technology and applications. Data are taken from BHRA's ten secondary abstract publications, which abstract over 550 technical journals as well as books, proceedings, standards, technical reports, and British patents. Major fields covered include civil engineering hydraulics, industrial aerodynamics, dredging, fluid flow, fluid power, fluid sealing, fluidics feedback, and tribology.

2. CIM - Inventory of Models

The Central Inventory of Models data base is maintained by ORNL for DOE, and includes energy-related bibliographic and numeric data bases, graphics packages, integrated hardware/software systems, and models from DOE laboratories.

3. COMPENDEX - Engineering

COMPENDEX covers significant world-wide engineering literature (1970 to date) from ~2,000 serials and over 900 monographic publications (including books and conference proceedings). Fields of engineering and related subject areas include: aerospace engineering, agricultural engineering and food technology, automotive engineering, bioengineering, chemical engineering, civil engineering, computers and data processing, construction materials, control engineering, electrical engineering, electronics and communications engineering, engineering geology, engineering physics, fluid flow, and heat and thermodynamics. Also covered are industrial and management applications, instruments and measurements, light and optical technology, marine engineering, material properties and testing, mechanical engineering, metallurgical and mining engineering, nuclear technology, ocean and underwater technology, petroleum engineering, railroad engineering, transportation, water and waterworks engineering, and pollution, sanitary engineering, and waste.

4. CONF - Conference Papers

This includes scientific and technical papers (1973 to date) in the life sciences, physical sciences, and engineering areas that are presented at regional, national, and international meetings, including small meetings having a cross-disciplinary focus.

5. EDB - Energy

DOE Energy is one of the world's largest sources of literature references on all apsects of energy and related topics. It includes references to journal articles, report literature, conference papers, books, patents, dissertations, and translations. All manner of energy topics are included: nuclear, wind, fossil, geothermal, tidal, etc., as well as the related topics of environment, policy, and conservation.

6. EIA - Energy Information

EIA Citations are drawn from Energy Information Abstracts, and are compiled by the Environmental Information Center.

7. FEDEX - Federal Government Activities

Federal Index contains information (1976 to date) on federal government activities drawn from the Congressional Record, the Federal Register, The Weekly Compilation of Presidential Documents Commerce Business Daily, and the Washington Post. Additional sources from the F & S Index are also included, beginning in 1979. The citations provide access to the Code of Federal Regulations, the U.S. Code, Public Laws, Congressional Bills, Resolutions and Reports. The information is indexed by acting government agency, affected industry, or institution and type of government action or function.

8. ISMEC - Mechanical Engineering

ISMEC covers mechanical engineering, production engineering, and engineering management. Subjects covered include production processes, tools and equipment, energy and power, transport and handling, management and production, measurement and control, and mechanics, materials and devices. References (1973 to present) ar gathered from journal articles, technical reports, conference proceedings, and books.

9. NSA - Nuclear Science

The Nuclear Science Abstracts base presently contains more than 500,000 citations, covering the period 1967 to June 1976.

10. NSC - Nuclear Safety

The nuclear safety information data base is maintained by the Nuclear Safety Information Center, ORNL, under the joint sponsorship of DOE and NRC.

11. NTIS - Government Sponsored Research

NTIS covers U.S.-government-sponsored research and development technical reports from over 200 Federal agencies and some reprints, federally-sponsored translations, and foreign-language reports in major areas of technical interest. Its multi-disciplinary scope includes aeronautics, agriculture, astronomy and astrophysics, atmospheric sciences, behavioral and social sciences, biological and medical sciences, chemistry, earth sciences and oceanography, electronics and electrical engineering, energy conversion (non-propulsive), materials, and mathematical sciences. Also covered are mechanical, industrial, civil, and marine engineering, methods and equipment, military sciences, missile technology, navigation, communications, detection methods and counter-measures, nuclear science and technology, ordnance, physics, propulsion and fuels, and space technology.

12. RSI - Radiation Shielding Information

The Radiation Shielding Information Center data base is maintained by ORNL and contains citations to literature describing computer codes that have been designed to perform radiation analysis and shielding calculations, neutron cross-section processing, and experimental data analysis.

13. WELDASEARCH - Joining of Metals and Plastics

The WELDASEARCH data base provides primary coverage of the international literature on all aspects of the joining of metals and plastics and related areas such as metals spraying and thermal cutting. WELDASEARCH includes material on welded design, welding metallurgy, and fatigue and fracture mechanics, as well as welding and joining equipment, corrosion, thermal cutting, and quality control. Approximately 5,000 new records are added to WELDASEARCH each year from several thousand journals and research reports, books, standards, patents, theses, and special publications.

APPENDIX B
OCONEE LICENSEE EVENT REPORTS (LERs)*

1. Cooldown Rate Limit Exceeded Following Loss of ICS Power at Oconee-3

The RCS cooldown rate limit was exceeded after power to the ICS was lost for about two and one-half minutes. No ES actuation setpoints were reached, and adequate RCS inventory was maintained. No damage was incurred. Loss of ICS power resulted from blown fuses in normal inverter (KI) and failure of transfer switch to transfer automatically to regulated AC power. When ICS power was restored, excessive feedwater flow caused a rapid RCS cooldown. A redundant transfer switch has since been installed, and personnel have been instructed on how to respond properly to loss of ICS power.

2. Reactor Coolant System Cooldown Rate Excessive at Oconee-3

During a routine shutdown for maintenance, a minor system transient occurred, which resulted in opening a power-actuated pressurizer relief valve with reactor power at 15%. The valve remained open and the RC system depressurizaton continued until the isolation valve was closed. The shutdown continued with a cooldown rate of 100°F/hr. However, when the initial drop in temperature from depressurization was included, the rate exceeded the 100°F/hr tech spec limit by 1°F/hr. It was determined that boric acid crystal buildup on the connecting pin of the lever arm of the pilot valve had caused the valve to remain open.

3. Additional Information on Excessive Cooldown Rate at Oconee-3

Reactor power was being reduced from 100% to 15% by the integrated control system for a maintenance shutdown. When 15% was reached, unit load demand was 65 MWe and power generation was 115 MWe. This difference existed because the reactor was operating at its lower limit of 15% and could not follow load demand. A transient occurred that tripped the reactor. During the transient, a relief valve opened and failed to close. This transient was terminated by closing the isolation valve. Cooldown rate was 101°F/hr during the first hour. The relief valve failed because of heat expansion, boric acid crystal buildup on the valve lever, and bending of the solenoid spring bracket.

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*Text has been modified slightly in some instances to improve clarity and readability.

4. Feedwater Transient Following Scram Actuates HPI at Oconee-1

On December 13, 1978, the T_{ave} statalarm began to act erratically and an investigation was initiated. During investigation (12/14/78) a power cord supplying T_{ave} recorder shorted, causing an apparent (not real) drop in T_{ave} of 13°F and ICS attempted to correct T_{ave}. Unit tripped on high pressure/temperature. Feedwater transients during cooldown allowed OTSG "B" to go dry. When it was refilled it caused RCS pressure to drop below 1500 psi, which actuated the HPIS. The cause of the T_{ave} cord short has not been identified. The feedwater transients were probably caused by improper valve operation. The power supply cord was replaced.

5. Reactor Coolant System Cooldown Rate Exceeds Limits at Oconee-2

When a spurious signal in the 230 kV switchyard circuit breaker failure relay circuitry resulted in the isolation of the switchyard, the reactor scrammed from 75% power. The scram tripped the feedwater pumps. The emergency feedwater pumps started and filled the steam generators to the 95% level as designed. This high water level, plus normal required steam, resulted in a cooldown rate of 140°F/hr in one loop and 135.5°F/hr in the other, which exceeds the 100°F/hr limit. Reduction in water level set point is being studied.

APPENDIX C

SPECIFIC DOCUMENT REFERENCES

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2. "Parametric Analysis of Rancho Seco Overcooling Accident," letter, R. Cheverton to M. Vagins dated March 3, 1981.

3. Effect of HPI on Vessel Integrity for Small Break LOCA Event with Extended Loss of Feedwater, BAW-1648 (November 1980).

4. "Runaway Feedwater After Turbine Trip Report," letter, M. Levine to N. Zuber dated July 2, 1980.

5. "Transmittal of Preliminary Calculations of a Steam Line Break Accident," letter, S. Fabic to C. Serpan dated May 14, 1981.

6. "Analysis of a Steam Line Break with Primary System Overcooling for a Typical B&W Reactor", letter, R. Cerbone to R. Kryter dated August 14, 1981.

7. "Completion of Scheduled Analyses on Pressurized Thermal Shock Scenarios," letter, S. Fabic to C. Serpan dated June 22, 1981.

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9. "Oconee Nuclear Station Docket Nos. 50-269, -270, -287," Letter Report, W. O. Parker, Jr. to H. R. Denton dated July 23, 1980.

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12. "Thermal Shock to Reactor Pressure Vessels," Letter Report, R. W. Jurgensen to D. G. Eisenhut dated May 14, 1981.

13. "Reactor Vessel Brittle Fracture," Letter Report, J. J. Mattimoe to H. R. Denton dated May 12, 1981.

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19. "IRT Output," letter, M. M. Levine to R. D. Cheverton dated August 11, 1980.

20. "IRT Results," letter, M. M. Levine to N. Zuber dated July 2, 1980.