UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312 SP
	)	
(Rancho Seco Nuclear Generating Station)	)	

AFFIDAVIT OF ROBERT A. CAPRA

I, Robert A. Capra, being duly sworn, depose and state that:

1. I am an employee of the U.S. Nuclear Regulatory Commission (NRC). My present position is Technical Assistant to the Director, Division of Systems Integration within the Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached to this affidavit.

2. The purpose of my affidavit is to respond to Item No. 4 of the Atomic Safety and Licensing Appeal Board Memorandum and Order dated October 7, 1981 (ALAB-655). Item No. 4 requests the following information:

   "Status reports from the staff and SMUD on the need for additional analyses identified in the Staff Evaluation at 19, 23 (see p. 16 supra)".

   Specifically, page 16 of ALAB-655 requests information on the following items:

   "(1) the more detailed small-break loss-of-coolant accident (LOCA) analyses discussed in Sections 8.4.1 and 8.4.2 of NUREG-0560, "Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company," and (2) analyses to (a) confirm that AFW, if lost, can be restored within a reasonable period of time and (b) describe the thermal-mechanical behavior of vessel materials under these conditions."

   Items (1) and (2) are addressed separately below.

3. <u>More Detailed Small-Break Loss-of-Coolant Analyses</u>

The Staff Evaluation referred to in ALAB-655 is entitled "Evaluation of Licensee's Compliance With the NRC Order Dated May 7, 1979," forwarded to the licensee by letter from Harold R. Denton to J. J. Mattimoe, dated June 27, 1979. Page 19 of the Staff Evaluation states:

> "To support longer term operation of the facility, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated transients. More detailed analyses of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Section 8.4.1 and 8.4.2 of the recent NRC "Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company" (NUREG-0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future."

Shortly after the accident at TMI-2, the Lessons Learned Task Force was formed. The purpose of the Task Force was to identify and evaluate those safety concerns originating with the TMI-2 accident that required licensing actions. Their charter included: the review and evaluation of investigative information; staff evaluations of responses to IE Bulletins and Orders; recommendations from the Commissioners, ACRS, outside organizations, and the NRC staff. The recommendations from NUREG-0560 were also specifically considered. The analyses discussed in Sections 8.4.1 and 8.4.2 of NUREG-0560 were redefined by the Lessons Learned Task Force and appear as Recommendation 2.1.9 of NUREG-0578, "TMI-2 Lessons Learned

Task Force Status Report and Short-Term Recommendations" (July, 1979). A detailed discussion of this item is found on pages A-42 through A-47 of NUREG-0578. Basically, Recommendation 2.1.9 required that licensees provide analyses, procedures and training that address three areas: (1) small-break LOCAs, (2) inadequate core cooling and, (3) transients and accidents. This recommendation was eventually incorporated into the TMI Action Plan (NUREG-0660) as item I.C.1 and is being evaluated on all operating reactors through the Action Plan's implementation document NUREG-0737, "Clarification of TMI Action Plan Requirements" (November, 1980).

In response to items (1) and (2) above, the licensee provided the required analyses and generic operator guidelines to address small-break LOCAs and inadequate core cooling in a letter from J. J. Mattimoe to R. W. Reid, dated November 26, 1979. The staff reviewed these analyses and guidelines and approved their use pending the eventual incorporation of both items into the broader program being addressed under item (3), "Transients and Accidents." To address this requirement, the Babcock and Wilcox Owners' Group, of which SMUD is a member, has embarked on an extensive program to upgrade the analyses, procedures and training for transients and accidents. The program, which is referred to as ATOG (Abnormal Transient Operating Guidelines), has among its objectives a realistic investigation of a wide range of reactor plant transients and accidents including small-break LOCAs. This investigation includes combinations of failures and operator actions not previously considered in licensing, and will thereby provide a source of readily available information for plant operators to: (1) deal more effectively with abnormal situations and (2) have a more thorough knowledge and understanding of plant response under these conditions. The program consists in part of coordinated event trees, safety sequence diagrams and analyses that will ultimately lead to symptom-oriented procedures for the operator. These procedures will replace the currently existing event-oriented procedures.

The B&W Owners' Group program does not include the development of guidelines generic to all B&W operating plants, but rather, it includes plant-specific guidelines. The guidelines are being developed for the plants in a sequence developed by the Owners' Group. The lead plant in the sequence is Arkansas Nuclear One, Unit One (ANO-1). A draft of the operating guidelines and supporting analyses for ANO-1 was submitted in August, 1980 and accepted by the staff as a formal submittal in December, 1980. Operating guidelines have also been submitted recently for the three Oconee units. SMUD, as well as B&W, who is performing the analyses and developing the guidelines, have stated that the plant-specific guidelines will differ only slightly from plant to plant and that the methods used to develop the guidelines are the same for each plant. The plant-specific guidelines for Rancho Seco are scheduled for submission by July 1, 1982 with conversion of the guidelines into plant procedures during the 1982 refueling outage (September, 1982). This schedule is consistent with the schedule for other operating reactors.

In addition to the discussion presented above, additional small-break LOCA analyses were required of SMUD and the other B&W licensees subsequent to the Staff Evaluation of June 27, 1979. The results of these analyses, as well as the staff's evaluation of the submittals, are documented in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed 177-FA Operating Plants," dated January, 1980. This document was presented as Staff Exhibit 2 during the original proceeding.

4. <u>Timely Restoration of Auxiliary Feedwater and the Thermal-Mechanical Report</u>

Page 23 of the Staff Evaluation of June 27, 1979 states:

> "We conclude that further reliability analyses are needed as part of the long-term requirements of the Order to confirm that AFW can be restored (if lost) in a reasonable period of time. B&W has agreed to provide a detailed thermal-mechanical report on the behavior of vessel materials for these extreme conditions, to be applicable generically to the Oconee class of plants, which includes Rancho Seco."

As part of the staff's approach to assuring long-term AFW system reliability for Rancho Seco and the other B&W operating facilities, the staff met with the B&W Owners' Group on July 19, 1979. During this meeting, it was suggested by the staff and agreed to by the Owners's Group that they would perform an AFW system reliability analysis on their facilities, using the same methodology as the staff used for the Westinghouse and Combustion Engineering plants. The staff met again with the Owners' Group on August 9, 1979 to discuss the detailed scope and schedule for completing the reliability analysis. Rancho Seco was chosen as the lead plant for this effort. On December 17, 1979, in a letter from J. J. Mattimoe to R. W. Reid, SMUD provided the results of its AFW system reliability analysis. Based upon the staff's review of this report, the staff met with SMUD on February 12, 1980 to discuss upgrade requirements for the Rancho Seco AFW system. At this meeting, the staff discussed with the licensee the results of the staff's review of the reliability analysis. In addition, the staff and licensee agreed that it would be desirable to combine all requirements associated with the long-term

AFW system upgrade into one staff evaluation. Thus on February 26, 1980, the staff issued its evaluation of the Rancho Seco AFW system and identified the required upgrades. This evaluation included requirements from four major areas:

(1) The long-term portion of the May 7, 1979 Order specifically related to improving AFW system timeliness and reliability;

(2) Improvements required as a result of the identification of dominant failure contributors identified in the Rancho Seco AFW reliability analysis;

(3) Requirements associated with automatic AFW initiation and flow indication identified in NUREG-0578; and,

(4) Additional requirements generic to most pressurized water reactors as identified by the staff during its review of AFW systems on Westinghouse and Combustion Engineering operating plants.

A detailed status regarding the licensee's implementation of the requirements identified in the staff's evaluation of February 26, 1980 is provided in response to item No. 2 of ALAB-655. The items identified in the February 26, 1980 staff evaluation were aimed at improving the timeliness and reliability of delivery of AFW system flow to the steam generators. Thus, the reliability analyses identified in the Staff Evaluation of June 27, 1979 have been completed by the licensee and evaluated by the staff.

With respect to the thermal-mechanical report, the TMI Action Plan (NUREG-0660) and NUREG-0737 include a task (II.K.2.13) that requires a detailed analysis for all pressurized water reactors (PWRs) of the potential for thermal shock of reactor vessels resulting from cold safety injection flow during small break LOCAs. A generic report entitled "Thermal-Mechanical Report - Effect of HPI on Vessel Integrity for Small Break LOCA Event With Extended Loss of Feedwater," (BAW-1648) was prepared by Babcock & Wilcox (B&W) in November, 1980 to address this requirement. The licensee reviewed this report and endorsed its applicability to Rancho Seco by letter from J. J. Mattimoe to R. W. Reid, dated January 16, 1981.

Subsequent to and apart from the report's submission, the staff met with the PWR industry Regulatory Response Groups (RRGs) and the PWR reactor manufacturers on March 31, 1981 to discuss the NRC's concerns related to the generic issue of "Pressurized Thermal Shock." The RRG representatives agreed to send the NRC a report by May 15, 1981, describing their ongoing work associated with the issue of pressurized thermal shock. At a progress briefing from the PWR Owners' Groups on April 29, 1981, the Owners' Group representatives asserted that there was no need for immediate corrective actions. These assertions were based on the low probability of severe overcooling transients, as well as the high fracture toughness of the vessel at this time. They agreed, however, to provide more technical backup in their May 15 report as well as provide a generic basis for continued safe operation of the plants.

On May 12, 1981 the B&W RRG provided its response to the March 31 and April 29 requests in a letter from J. J. Mattimoe to H. Denton (Enclosure 1). This report concluded that the small break LOCA transient analyzed in the Thermal-Mechanical Report (BAW-1648) was the bounding transient for B&W operating reactors with respect to the limiting overcooling transient that should be considered for reactor vessel brittle fracture. Also on May 12, 1981, the Atomic Safety and Licensing Board for Rancho Seco was sent a Board Notification from D. G. Eisenhut entitled "Board Notification - Thermal Shock to PWR Reactor Pressure Vessels," (Enclosure 2). Included in the Board Notification was a preliminary assessment of the thermal shock issue and the staff's basis for allowing continued operation of licensed PWRs.

In a letter from J. J. Mattimoe to D. G. Eisenhut, dated May 22, 1981 (Enclosure 3), SMUD stated that it had determined that the information provided in the B&W RRG report of May 12, 1981 was applicable to Rancho Seco and that SMUD supported the conclusions of the report. The letter also provided additional comments regarding the analysis presented in BAW-1648 as well as SMUD's commitment to perform further plant-specific analyses to demonstrate that considerable time exists before there are any concerns over brittle fracture during such an event at Rancho Seco.

Based on the discussion presented above, it is obvious that there is a direct tie between BAW-1648 and the pressurized thermal shock issue. Therefore, the review of the Thermal-Mechanical Report has been incorporated into the Task Action Plan for resolving the Pressurized Thermal Shock issue. Two additional documents directly related to this issue are attached for the information of the ASLAB: (1) SECY-81-286A, entitled "Pressurized Thermal Shock of Pressure Vessels," dated September 8, 1981 (Enclosure 4) and (2) Memorandum for the Commissioners from William J. Dircks, entitled "Staff Review of ORNL Report on Pressurized Thermal Shock," dated October 30, 1981 (Enclosure 5). Enclosure 4 discusses recent staff actions concerning the issue of pressurized thermal shock and outlines the present plan for dealing with the issue on operating plants. The B&W Owners' Group program includes plant-specific analyses that will utilize known plant data and realistic assumptions instead of the bounding and overly conservative assumptions utilized in BAW-1648. These analyses will be completed for the lead plant (Oconee 1) by December 31, 1981 and for Rancho Seco by July 1, 1982. Enclosure 5 provides the staff's most recent assessment of the acceptability of continued operation of PWRs considering the pressurized thermal shock issue. A principal conclusion stated in Enclosure 5 is that an overcooling event similar to the most severe transient that has occurred (the Rancho Seco event of March 20, 1978) will not pose a threat to the Oconee-1 pressure vessel for several more years. (The Oconee 1 pressure vessel is similar to that of Rancho Seco.) Another important conclusion stated in Enclosure 5 is that if certain events more severe than the Rancho Seco overcooling event were to occur today, and if the reactor pressure vessel were to remain at high pressure or be repressurized, and if fracture mechanics calculations believed to be conservative are used, then vessel failure may be predicted for the Oconee 1 vessel. However, the staff had previously concluded, and discussed with the Commission, that the probability of occurrence of pressurized overcooling events more severe than the Rancho Seco event is sufficiently low that immediate corrective action is not warranted.

5. In summary, the staff believes that the analyses identified in its Staff Evaluation of June 27, 1979, as necessary for long-term operation have: (1) been required of the licensee; (2) been submitted by the licensee (with the exception of the ongoing ATOG program); and (3) been reviewed and evaluated by the staff (with the exception

of the ATOG program for Rancho Seco and the Thermal-Mechanical Report). However, where this information has either not been submitted and/or evaluated by the staff, ongoing programs are underway to assure their completion.

6. The above statements and opinions are true and correct to the best of my personal knowledge and belief.

Robert A. Capra
Robert A. Capra

Subscribed and sworn to before
me this 11th day of December 1981

Judy L. Butts, Notary Public
My Commission Expires: July 1, 1982

Robert A. Capra
STATEMENT OF PROFESSIONAL QUALIFICATIONS

Since June 1980, I have served as Technical Assistant to the Director, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC). In this capacity, I work for the Division Director on various technical and administrative matters associated with review and regulation of operating nuclear power plants and plants under review for operating licenses.

I enlisted in the United States Navy in July 1964 and served in that capacity for three years. During that time my duties included attending the Enlisted Naval Nuclear Power School, Mare Island, California followed by subsequent study and qualification as a reactor operator and staff instructor on the Navy's "D1G" reactor located in West Milton,New York.

Following my enlistment, I attended the United States Naval Academy where I graduated in June 1971 with a Bachelor of Science degree in Marine Engineering and was commissioned as a line officer in the United States Navy. Additional graduate level studies in nuclear reactor theory, thermodynamics, electrical engineering, health physics and other related engineering fields were completed in 1972 at the Officer Naval Nuclear Power School, Bainbridge, Maryland. I subsequently returned to West Milton, New York where I studied and qualified as Engineering Officer of the Watch on the Navy's "D1G" reactor.

From 1973 to 1976, I served aboard an operating nuclear submarine, during which time my duties included standing watch as Engineering Officer of the Watch and directing, training and supervising personnel under my supervision in the operation, maintenance and repair of various equipment and systems primarily associated with the ship's nuclear reactor. During this period my assignments included supervision of the Operation Department, Electrical Division, Reactor Controls Division, Main Propulsion Division and the Chemistry and Radiological Control personnel. In addition, I qualified as Chief Engineer for the supervision of operation and maintenance of Naval Nuclear Propulsion Plants.

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Enclosure 1

SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

May 12, 1981

RECEIVED
MAY 15 1981
U.S. NUCLEAR REGULATORY COMMISSION

DR HAROLD DENTON DIRECTOR
OFFICE OF NUCLEAR REACTOR REGULATION
U S NUCLEAR REGULATORY COMMISSION
WASHINGTON D C 20555

DOCKET 50-312, RANCHO SECO NUCLEAR GENERATING STATION, UNIT NUMBER ONE
REACTOR VESSEL BRITTLE FRACTURE

At your request, we met with the NRC on March 31, 1981. At that meeting, your staff presented information on overcooling transients with repressurization. As a result of that meeting, we agreed to present a letter report by May 15, 1981 to put the thermal shock issue into perspective. Again, on April 29, 1981 your staff requested another meeting in which additional information was requested to be presented in the May 15th report. This additional request included justification for continued operation. The attached letter report provides the requested information. This report has been discussed with representatives of the utilities with Babcock & Wilcox operating plants, and they concur with the conclusions. Each Babcock & Wilcox licensee plans to make a submittal by May 22, 1981 which will identify the planned plant specific actions for thermal shock.

At the March 31, 1981 meeting, we agreed to furnish you with a review of the current industry research efforts in the area of reactor vessel thermal shock. Attachment two is a summary of the ongoing programs by the Electric Power Research Institute.

John J. Mattimoe

John J. Mattimoe
Chairman of the Babcock & Wilcox
Regulatory Response Group

Attachments: 1) Letter report on Reactor Vessel Brittle Fracture concerns in Babcock & Wilcox Operating Plants

2) Summary of the Electric Power Research Institute Programs pertaining to brittle fracture

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cc: G. P. Beatty, Jr. (FPC); K. S. Canady (DPCO); D. C. Trimble, (AP&L); R. F. Wilson (GPU); and W. C. Rowles (TECO)

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LETTER REPORT ON REACTOR VESSEL

BRITTLE FRACTURE CONCERNS

IN B&W OPERATING PLANTS

Document Identifier
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May 15, 1981

Prepared by:
Babcock & Wilcox Company for the
Owners Group of Babcock and Wilcox
177 Fuel Assembly NSS Systems

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Reactor Vessel Brittle Fracture

I. Abstract:

This letter report summarizes the evaluations made to date regarding possible brittle fracture of B&W operating plant reactor vessels during transients that result in severe overcooling with potential repressurization of the reactor vessel. It was prepared in response to an NRC request during a March 31, 1981 meeting between the NRC and various industry groups. The basis for concluding that there is no immediate brittle fracture concern (into 1983) for B&W operating units resulting from thermal shocking of the reactor vessel during small break LOCA transients is presented. A comparison of the small break LOCA event with other overcooling events is made to demonstrate the small break analysis bounds the overcooling transient. Long term plans to resolve the concern are summarized.

II. General:

A. Reactor Vessel Brittle Fracture during Design Basis LOCA

Babcock & Wilcox evaluated the capability of its pressurized water reactor vessels to withstand thermal shock caused by the double-ended rupture of a 36-inch-diameter hot leg pipe as early as 1969.(1)
At that time, the hot leg rupture was ascertained to represent the most severe LOCA condition (i.e. from the standpoint of a brittle fracture failure). Based on this early analysis of the hot leg rupture. it was concluded that "The reactor vessel will not lose its integrity due to crack propagation as a result of thermal shock caused by actuation of the ECCS following a LOCA even if this transient occurs at the end of 40 years of irradiation and the vessel wall contains a flaw of critical size".

B. Reactor Vessel Brittle Fracture during Small Break LOCA

As a result of the TMI-2 transient, new operating guidelines were issued which included operation of the HPI system in a once-thru cooling mode as a means of core cooling until the plant could be cooled and depressurized and then placed on the decay heat system. This mode of operation raised new questions concerning the thermal shocking of the reactor vessel due to the cold HPI flow being inj[illegible]nto the vessel with no RCS flow.

Because of these new considerations and in response to NUREG-0737[2], analyses were performed in 1980 for the small break LOCA transients with extended loss of feedwater. Reports documenting these analyses were submitted to the NRC by the Licensees in January, 1981.[3],[4]

Recently, the issue has been raised by the NRC as to whether or not the small break loss-of-coolant transient with extended total loss of feedwater indeed represents the worst overcooling transient which should be considered with regards to reactor vessel brittle fracture. This report addresses this concern and concludes that the small break LOCA transient (as analyzed in BAW-1648) is the limiting transient for the B&W NSSS designs. This limiting event is, therefore, treated in some detail in the following section, followed by sections discussing the Non-LOCA events, other activities (ongoing and planned) related to the brittle fracture concern and finally a summary presenting justification for continued plant operation.

Small Break LOCA - Specific

The small break LOCA transient with extended loss of feedwater has been thoroughly analyzed with regard to reactor vessel brittle fracture[3],[4]. (A description of the transient scenario is provided in Section 1 of Reference 3.) The analyses envelope all of the B&W operating units, (i.e., worst-case inputs are combined). Some of the salient conservative assumptions used in these generic analyses are as follows:

1. All feedwater is lost for an extended period of time.
2. All reactor coolant flow is lost for an extended period of time.
3. Core flow into the downcomer is assumed to pass through four vent valves rather than the eight valves existing on all but one plant. This reduces the amount of warm water entering the downcomer.
4. A hypothetical maximum HPI flow capacity is assumed over the entire RCS pressure range analyzed. No single plant can achieve this hypothetical capacity over the entire pressure range. This assumption affects all the analyses, including those which assume operator action to throttle HPI, since the initial reactor vessel cooldown prior to achieving 100°F subcooled conditions at the core outlet is maximized, resulting in increased thermal stress during the transient.
5. A worst-case HPI fluid temperature of 40°F was assumed.
6. Linear elastic fracture mechanics (LEFM) methods were used in the brittle fracture analysis. No credit was taken for warm prestressing.
7. Materials information was taken from Regulatory Guide 1.99.
8. Reactor vessel most limiting welds were assumed to be located directly beneath the cold leg inlet nozzles.
9. Reactor vessel cooldown was calculated based on a one-dimensional heat conduction analysis.
10. Mixing in the cold leg piping was not modeled.

The major uncertainty associated with the analyses is the degree of heatup of the high pressure injection water due to

- Upstream mixing in the cold leg piping
- Heating by the reactor vessel walls
- HPI pump energy (minimal)
- Heating by the cold leg piping (minimal)
- Mixing with vent valve fluid

The last item, the preheating of the incoming HPI by mixing with vent valve fluid, represents the most significant contributor to reducing the brittle fracture concern.

In order to evaluate the thermal shock concern, various thermal hydraulic assumptions were made. The major thermal hydraulic assumptions were:

1. <u>Bounding Assumptions</u>

   Analyses were performed assuming no heatup of HPI due to any of the above effects. When natural circulation was assumed to be inhibited at approximately 10 min. into the transient, the downcomer fluid temperature at reactor vessel wall was ramped to the BWST temperature ($40^{o}F$ or $90^{o}F$) in approximately 60 seconds. This case is essentially a zero mixing case after 10 minutes into the transient.

2. <u>Mix Assumptions</u>

   Analyses were also performed assuming HPI fluid enters the downcomer, mixes with the warmer vent valve flow, which is assumed to be circumferentially distributed, and then streams down the reactor vessel wall. This is believed to be a more realistic assumption since some degree of HPI mixing and heatup is expected.

Also, the reactor vessel fluences were obtained from the Effective Full Power Years (EFPY) determined from core follow and the methodology as outlined in BAW-1511P which was submitted to the NRC on March 12, 1981. This document represents a significant effort as part of the B&W Owners Group since 1976.[9]

The EFPY on B&W operating plants as of 4/27/81 is as follows:

Rancho Seco	3.45 EFPY
Oconee I	4.90 EFPY
Oconee II	4.36 EFPY
Oconee III	4.21 EFPY
Crystal River III	2.19 EFPY
TMI-1	3.52 EFPY
Davis Besse-I	1.25 EFPY
Arkansas Nuclear One Unit 1	3.91 EFPY

BAW-1511P also contains information on Quality Assurance of Reactor Vessel weld properties. This includes weld number, vessel in which located, type of filler wire, type of weldment and various other surveillance capsule measured and predicted information.

The analyses in BAW-1648 assumed operator action to throttle high pressure injection such that core outlet conditions would be maintained less than 100°F subcooled. Appropriate revisions to the Small Break Operating Guidelines have been issued to the affected Utilities. In addition, B&W has recommended to the operating plants that BWST temperatures be maintained greater than the Technical Specification minimum of 40°F.

The conservative bounding assumptions were used in the 1980 generic analyses(3),(4) with the intent being to define the extent of the brittle fracture problem. With these conservatisms, the following conclusions resulted from the analyses:

1. Rancho Seco and Oconee I reactor vessels represent the most and the second-most limiting operating B&W units respectively at this point in time. The limiting welds, as analyzed, with respect to brittle fracture in these reactor vessels are longitudinal welds. These vessels have limiting longitudinal welds near the cold leg nozzles. Hence, the analysis of these operating vessels currently bounds all others.
2. Using the conservative bounding thermal-hydraulic assumption (thermal hydraulic assumption #1 on page 4) plus combining worst case inputs in the generic analyses showed no immediate brittle fracture concern exists for the operating plants. The analyses show that operator action to throttle HPI flow will preclude brittle fracture.
3. Using the more realistic mix assumption (thermal-hydraulic assumption #2 on page 4) indicates the most limiting reactor vessel has more than one additional effective full power year beyond the present bounding analysis (i.e. into 1983) before any concern is approached, even considering worst-case BWST temperatures. This is illustrated in Figure 1, which shows allowable and actual pressures during the transient for the generic analysis using Rancho Seco weld material properties at 4.8 EFPY, assuming worst-case 40°F BWST water.(3) The actual Rancho Seco EFPY as of April 27, 1981 was 3.45 EFPY.

Therefore, given operator action to throttle HPI there is no immediate brittle fracture concern for B&W operating units resulting from thermal shocking of the reactor vessel during small break LOCA transients.

III Non-LOCA Overcooling Events

NUREG-0737, Item II.K.2.13, required that small break LOCA with extended loss of feedwater events be analyzed for reactor vessel brittle fracture. Recently, the ACRS and the NRC have expressed the concern that perhaps other transients, such as steam line breaks, which have the potential for overcooling and subsequent system repressurization, may be more limiting transients with respect to the reactor vessel brittle fracture concern.

As a result of the NRC's request in 1975 (Reference 5), our position regarding these repressurization events has been that operator action to mitigate system repressurization (by throttling HPI and utilizing atmospheric dump or turbine bypass valves) is adequate to keep reactor coolant pressure and temperature within technical specification limits over the service life of the reactor vessel.(6)

Table 1 compares primary system response during various overcooling events. As can be seen, the small break LOCA cases (case 1 and 2) already considered in BAW-1648 result in more overcooling (to approximately 90°F downcomer temperature) of the reactor vessel than unmitigated large steam line breaks.(7) Also, case 1, Table 1, clearly bounds all overcooling transients presented in Table 1 (with respect to the temperature transient). Based on these considerations, plus reliance upon the operator to mitigate the repressurization, the previous SBLOCA analyses are limiting, with respect to the brittle fracture concern. Assessment of the non-LOCA overcooling events (including subsequent repressurization) has confirmed this for operation into 1983.

IV. Concerns Expressed in Basdekas' letter to Udall 4/10/81

The Basdekas' letter of 4/10/81 has been reviewed and clarifications of several items for B&W designed plants are provided below. The quoted sentences have been extracted from the letter

A. "Such transients can cause the reactor vessel to cool-down to about 150°F in about 15 minutes, while the ECCS repressurizes it to about 2400 PSI."

In response to the IE Bulletin 79-05C, and as indicated in Section III, a large steam line break was analyzed. The analysis assumed both OTSG's blowdown, no Main Steam Isolation Valve (MSIV) closure and Emergency Feedwater at full capacity. The results indicate a minimum Reactor Coolant System (RCS) temperature of 230°F will be reached approximately 14 minutes into the transient.(7) Operator actions to throttle HPI flow will prevent repressurization of the RCS to 2400 PSIG.

B. "A reactor vessel fracture is one of the most serious accidents a reactor may experience. Depending on its location and mode, it is almost certain that it will cause a core meltdown with all its public health and safety ramifications, on which, I am sure, I need not elaborate for you."

It is very unlikely that a reactor vessel fracture, at a location and mode which results in a core meltdown, will occur. This is demonstrated by the positive margins resulting from analyses previously performed.(1,3,4)

C. "This is supported by analyses performed for the NRC. indicating that the overcooling transient that took place at Rancho Seco on March 20. 1978 would have caused such a vessel to rupture, had it been in operation for about 10 FPYE."

We are not aware of the information that Mr. Basdekas has, but the Rancho Seco vessel on March 20, 1978 had only 1.55 EFPY of irradiation and therefore appreciable margin for Brittle Fracture at that time. In an analysis prepared for the NRC by Oak Ridge National Laboratory (ORNL to Mr. Milton Vagins (NRC) dated March 3, 1981) a different analysis (Warm Prestressing) than that the one used in BAW-1648 indicates that the Rancho Seco Vessel has a useful Full-Power Life greater than 14 EFPY.

D. "Furthermore. a recent discovery of a discrepancy existing between the estimated vs. the measured values of neutron fluence for the Maine Yankee reactor vessel indicates a generic problem that makes things worse. The results of dosimetry measurements indicate the actual neutron fluence to be some 2.3 times higher than that estimated in the Maine Yankee Final Analysis Report."

The fluence discrepancy at Main Yankee was apparently due to lack of azimuthal flux variation in their calculational model and/or the use of cycle 1 extrapolated data. Azimuthal variations in a B&W reactor are on the order of a factor of 2 from maximum to minimum. Core escape flux is generally lower during cycle 1 (compared to subsequent cycles), and, therefore, ex-core fluences would be low. The fluence analysis procedure used at B&W accounts for azimuthal flux variation by using the two-dimensional transport code DOT to model reactor and surveillance capsules: and predicted fluences for extrapolated burnups are based on core escape flux from fuel management studies (PDQ criticality calculations) of future fuel cycles. B&W has always used the two-dimensional modeling approach whereas the initial Maine Yankee data were from a one-dimensional model.

The B&W procedure has been used to calculate the fluence exposure of capsules from five 177 FA reactors, four after cycle 1 and one after cycle 2 Comparisons to measured activities from capsule contained dosimeters have been $\pm 15\%$. All calculated data are subsequently normalized to dosimeter measurements before pressure vessel fluence is determined. These data are documented in BAW reports that are sent to the appropriate utility after each capsule is analyzed.

The B&W procedure was benchmarked when B&W participated in the "Blind Test" phase of the LWR Pressure Vessel Surveillance Dosimetry Program, an on-going study of surveillance analysis procedures that is operated by HEDL and ORNL for the NRC. B&W calculated fast flux as documented in NUREG/CR-1872, "Reactor Calculation Benchmarks - PCA Blind Test Results," January 1981, was within 10% of experimentally derived values at the simulated T/4 pressure vessel location in two experimental configurations. The "Blind Test" results are being documented in a NUREG report, but data are not identified with respect to participant.

E. "Moreover, as you may recall, one of the measures ordered by the NRC after the TMI-2 accident was to have all reactor operators not turn off the ECCS once it had been initiated."

Revised Small-Break LOCA Operating Guidelines have been issued to affected Utilities by B&W. The guidelines provide operator instruction on when to throttle the HPI flow to prevent repressurization.

V. Other Actions - The thermal shock concern has been addressed and programs have been either completed, currently underway or planned to assure safe operation.

A. Completed

1. ICS/NNI upgrades per IE Bulletin 79-27 and associated Commission orders.
2. EFW Systems Upgrades
3. Revised Small-Break LOCA Operating Guidelines regarding thermal shock have been issued to affected utilities.

   These guidelines are intended to:

   - Enhance understanding.
   - Provide operator instruction for HPI throttling on subcooling when in the HPI cooling mode with no RCS flow.
   - Emphasize re-establishing RC Loop Flow.
4. Abnormal Transient Operating Guidelines (ATOG) procedure under development to address item I.C.1 in NUREG-0737 include consideration of the brittle fracture concern.
5. B&W has recommended that Utilities maintain BWST temperatures higher than Technical Specification minimums.
6. BAW-1511P (reference 9) has been completed as part of an Owners Group program on Reactor Vessel materials.

B. Currently Underway

1. The Owners Group reactor vessel materials program is geared toward demonstrating adequate structural integrity of the reactor vessel throughout plant design life. Efforts currently underway include:
   - a determination of fracture toughness properties which are expected to demonstrate higher resistance to fracture than current industry predictions based on Charpy V notch specimens.
   - the development of less conservative fracture analysis procedures, which include elastic-plastic techniques.
2. Reactor Vessel Material Surveillance Programs in accordance with Appendix H of 10CFR50.

C. Immediate Future Plans

1. Plant specific evaluations to address the conservatisms associated with generic analyses are being investigated
2. More sophisticated vessel cooldown calculations are being considered to reduce the conservatisms associated with the one-dimensional heat conduction analysis previously employed.
3. Consideration of analysis for Non-LOCA events.

D. Long Term Plans

1. Discussions are in progress with EPRI regarding possible testing to obtain a better understanding of the thermal-hydraulic mixing phenomena associated with these overcooling transients.
2. CREARE, Inc. and other consultants have been contacted and involved in discussions concerning the thermal-hydraulic mixing aspects of the problems.
3. The investigation of enhanced inservice inspections methods with the objective being the reliable detection of smaller flaw sizes.
4. The evaluation of the in-place reactor vessel thermal annealing to recover some of the material properties lost through neutron irradiation.
5. The investigation of improved dosimetry and fluence calculations.

VI Summary - Justification for Continued Operation

As a result of the NRC's request of March 31, 1981 to put the reactor vessel brittle fracture issue in perspective, the following have been concluded:

A. Assessment of overcooling events indicates that the small break LOCA event as analyzed is bounding.
B. Generic analyses (including mixing) of the small break LOCA events show no immediate problem (into 1983) given operator action.
C. Revised operator guidelines have been issued. Immediate operator action is not required. Required operator action is straightforward.
D. Efforts are underway to resolve the long term issue.

Table 1

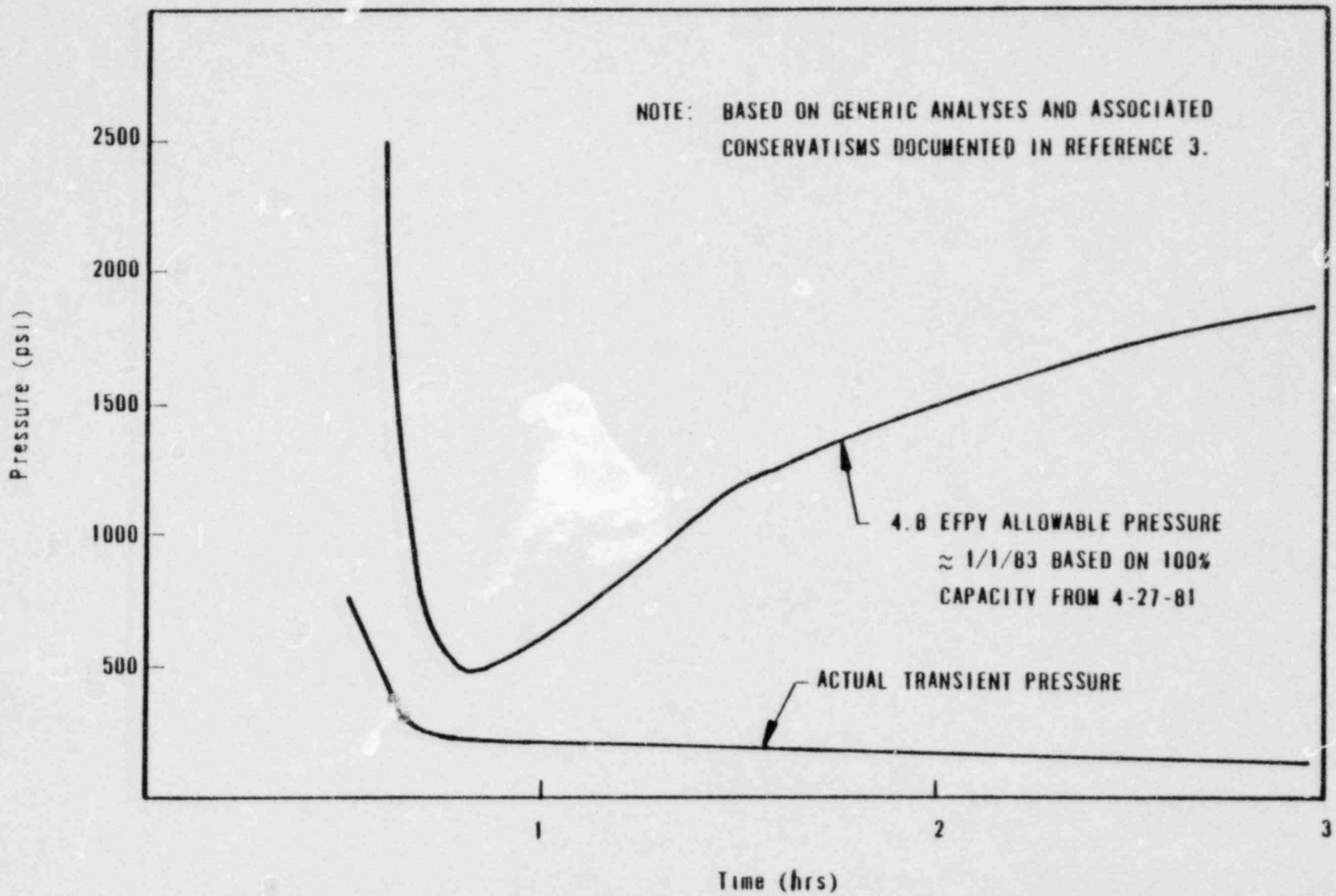
Primary System Response During Overcooling Transients

Case	Description	Cooldown Rate	Minimum Downcomer Temperature	Comment
1	BAW 1648 Bounding Analysis(3)	460F in 60 Seconds (460°F/min)	90F	No temperature recovery No repressurization* (90F BWST)
2	BAW 1648 Mix Analysis(3)	445F in 40 minutes (11.1°F/min)	90F	No temperature recovery No repressurization* (40F BWST)
3	Unmitigated Large Steam Line Rupture(7)	320F in 10 Minutes (32°F/min)	230F	Temperature recovers System repressurizes**
4	Rancho Seco Rapid Cooldown Incident of 3/20/78(8)	310F in 60 Minutes (5.2°F/min)	285F	Some temperature recovery Stable pressure between 1400 and 2100 psig

* Assuming operator action

** Can be mitigated by operator action

Figure 1 ALLOWABLE AND ACTUAL PRESSURES VS TIME, 0.023-FT^2 PRESSURIZER BREAK WITH OPERATOR ACTION, RANCHO SECO, 40F BWST, MIX2



References

(1) Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock, BAW-10018, Babcock & Wilcox, Lynchburg, Virginia, May 1969, Transmittal Letter, J. H. MacMillan (B&W) to Dr. P. A. Morris (AEC). Dated May 22, 1969.

(2) NUREG-0737, Item II.K.2.13

(3) Thermal-Mechanical Report - Effect of HPI on Vessel Integrity for Small Break LOCA Event with Extended Loss of Feedwater, BAW-1648, Babcock & Wilcox, Lynchburg, Virginia, November 1980.

(4) Reactor Vessel Brittle Fracture Analysis During Small Break LOCA Events with Extended Loss of Feedwater, BAW-1628, Babcock & Wilcox, Lynchburg, Virginia, December 1980.

(5) F. Schroeder (NRR) to K. E. Suhrke (B&W), Letter dated June 10, 1975.

(6) K. E. Suhrke (B&W) to F. Schroeder (NRR), Letter dated August 12, 1975.

(7) J. J. Mattimoe (SMUD) to R. H. Engelken (NRC), October 24, 1979.

(8) Committee Report on Rancho Seco Unit I Transient of March 20, 1978, Dated June 19, 1978, by Sacramento Municipal Utility District.

(9) Irradiation-Induced Reduction in Charpy Upper-Shelf Energy of Reactor Vessel Welds, BAW-1511P (Proprietary), Babcock & Wilcox, Lynchburg, Virginia, October 1980, Transmittal Letter, J. H. Taylor (B&W) to J. S. Berggen (NRC), Dated March 12, 1981.

EPRI RESEARCH ON THE PROPERTIES OF IRRADIATED MATERIALS PERTINENT TO THE OVERCOOLING TRANSIENTS

by

T. U. Marston

April 1981

The significance of an overcooling transient (OT) on the integrity of reactor[1] pressure vessels is related to many factors, but the principal contributors are the thermal stresses associated with the transient and the degree of radiation embrittlement. This memorandum reviews briefly the irradiated materials research sponsored by the Electric Power Research Institute (EPRI) germaine to the OT issue. The work is divided into four areas: the prediction of radiation damage, the measurement of fracture toughness of irradiated materials, the measurement of crack arrest conditions for irradiated materials, and thermal anneal treatment of embrittled reactor vessels. The radiation damage prediction research is significant to the OT issue because it helps determine what reactor vessel materials and therefore which power plants are of interest and defines the period of plant life for which them transients are of negligible importance. The fracture toughness research should help define what combinations of flaw size and transient severity should be of concern. The crack arrest research helps define the behavior of a given defect during and after the transient. The last research area, thermal annealing, is pertinent because this is the only method considered for the mitigation of radiation embrittlement. Each of the areas are to be discussed below in greater detail. A one-page summary of each research project is appended to this memorandum. A paper summarizing the thermal anneal research is attached for further information.

Radiation Damage Predictions

The OT need be only considered when the radiation embrittlement is sufficient to place the reactor vessel materials into the less tough, transition temperature range during the transient. Radiation exposure tends to increase the temperatures at which ferritic materials undergo a precipitous increase in fracture toughness (fracture resistance). This temperature is referred to as the NDT or RT_{NDT} or FATT. The degree to which a material is affected by radiation is a function not only of the total exposure (fluence) but the chemical composition and microstructure of the material. The currently recognized predictive methodology for radiation damage is the NRC Regulatory Guide 1.99.1 (ref. 1). The 1.99.1 relationships are bounding in philosophy therefore are not accurate predictions for many materials (ref. 2, 3). In the area of prediction development, EPRI is currently sponsoring four projects, RP1021-3, RP1240-1, RP1553-1, and RP1553-2. The focus is on the treatment of surveillance data accounting for the possible contributing effects of thermal aging, in situ annealing and alloy as well as trace elmental compositions. The work in this area should be completed by the third quarter of 1981. The end result is hoped to be an accurate predictive methodology for radiation damage.

Fracture Toughness of Irradiated Materials

The prediction of reactor vessel integrity relies on the use of fracture mechanics that integrates the effects of the component stresses, the flaws present, and the materials' resistance to fracture. With it use, the critical combinations of the three variables can be defined. The data base of fracture toughness (resistance to fracture) for irradiated materials is very sparse. Research projects 886-1 and 886-2 are designed to develop fracture toughness data on a range of reactor pressure vessel materials at increasing exposure levels. The materials included in the program represent weldments (principal focus), plates and forgings with the emphasis on the more embrittlement sensitive chemistries. The projects develop toughness data throughout the transition and into the upper shelf temperature ranges. The results should be directly applicable to the OT investigation. The relationship between the toughness indicated by the Charpy impact test and the fracture mechanics test is to be developed and/or verified. The research is scheduled to be completed by August of 1981.

Crack Arrest Conditions for Irradiated Materials

The research described above should help define the necessary conditions for the initiation of cracking but the conditions for the arrest of cracking under accident situations are also important. The material property necessary to define the conditions (crack size, stress level, etc.) for the termination of cracking is called crack arrest toughness. Research project 1326-1 is developing crack arrest data on irradiated materials. The materials are removed from actual reactor vessels and include two weldments and two plates representing both old and new fabricating and processing procedures. These test results will be used to verify or modify the ASME code procedure for predicting crack arrest toughness from Charpy data. The results may affect significantly the final depths of cracks following a transient and the potential for rupture if the vessel is repressurized. The first data are being generated and the testing will continue into 1982.

Thermal Anneal of Reactor Vessels

If the radiation embrittlement becomes too severe to insure vessel integrity, the only currently acknowledged procedure to reduce the accrued damage is thermal annealing. The thermal anneal procedure requires the vessel to be heated above its normal operating temperature by a margin of 100°F to 300°F to permit the bake out of the radiation induced damage. Although this technique has been demonstrated for a very small military reactor, it is untried on commercial scale vessels. Research project 1021-1 is designed to determine the feasibility of and develop the methodology for annealing an embrittled reactor vessel. The project is divided into two segments: the development of annealing treatments and the assessments of the practicality of those treatments. The first segment includes the multiple irradiation of three embrittlement sensitive weldments to assess not only the initial damage and subsequent recovery, but the re-exposure sensitivity and the recovery following the re-exposure. The second phase determines the procedures and evaluates the practicality and any possible detrimental side effects of the procedures. It has already been determined that the so called "wet anneal" ($T \leq 650°F$) is feasible, but may not be practical because of limited recovery (especially in transition temperature shift), required removal of the reactor fuel and lack of heat rejection capabilities. This research is to be completed by July 1981.

Summary

The memorandum reviews briefly the research on irradiated materials at EPRI that are pertinent to the OT issue. The bulk of the work should be completed by the end of 1981. Although the preliminary results indicate some relief can be gained from improved material condition predictions, it appears prudent to avoid the conditions, if at all possible, that make the OT an issue to begin with. That is the severe, or perceived to be severe, thermal down transients followed by significant repressurization.

RP1021-3 Steady State Radiation Embrittlement of Reactor Vessels

Prime Contractor: Westinghouse Electric Corporation (T. R. Mager)
Duration: March 1, 1979 to May 30, 1981 (to be extended to March 1982)
Project Cost: $948,947
EPRI Project Manager: T. U. Marston

Objective: The objective is to verify that irradiation embrittlement in reactor pressure vessels reaches a steady-state or saturation level, and to provide the knowledge needed for revising and/or developing new irradiation damage trend curves.

Strategy: Through testing of reactor vessel surveillance capsules, analysis of test data, and microstructural and mechanistic studies, verify the consistency of a "steady-state" irradiation embrittlement which has been observed and determine the mechanisms involved in radiation damage saturation in pressure vessel steels.

Status: Nine capsules have been received, six capsules have been tested and two surveillance reports have been prepared. Extensive metallurgical and ultrastructural investigations have been performed that indicate carbide alteration to be the principal embrittlement mechanism.

S. E. Yanichko et al., "Analysis of Capsule R From the Wisconsin Electric Power Company Point Beach Nuclear Plant Unit No. 2 Reactor Vessel Radiation Surveillance Program (WCA 9635)" EPRI RP1021-3 Topical Report, December 1979.

S. E. Yanichko et al., "Analysis of Capsule 12S From the Consumers Power Company Big Rock Point Nuclear Plant Reactor Vessel Radiation Surveillance Program (WCAP-9794)" EPRI RP1021-3 Topical Report, September 1980.

First Semiannual Technical Progress Report, October 1979.

Second Semiannual Technical Progress Report, April 1980.

Third Semiannual Technical Progress Report, October 1980.

T. R. Mager, "Analysis of Capsule R From the Wisconsin Public Service Corporation Kewaunee Nuclear Plant Reactor Vessel Radiation Surveillance Program (WCAP-9878)" EPRI RP1021-3 Topical Report, March 1981.

T. R. Mager, "Analysis of the Maine Yankee Reactor Vessel Second Accelerated Surveillance Capsule (WCAP-9875)" EPRI RP1021-3 Topical Report, March 1981.

RP1240-1 Collection and Evaluation of Data for Irradiated Pressure Vessel Steels

Prime Contractor: Fracture Control Corporation (R. A. Wullaert)

Duration: September 1, 1978 to August 31, 1981

Project Cost: $223,042

EPRI Project Manager: T. U. Marston

Objective: The primary goals of this program are to obtain and qualify the existing data on mechanical properties and irradiation history for irradiated pressure vessel steels employed in commercial light water nuclear reactors, and to develop a weighting procedure whereby the "quality" of the data can be systematically assessed. Finally a methodology for estimating the irradiated fracture toughness including K_{Ic}, J_{Ic} and J-R is to be developed and verified.

Strategy: All available surveillance data are to be assembled and qualified. Statistical analysis of the data using state-of-the-art modeling and statistical techniques will be made to develop embrittlement trends. Finally fracture toughness properties are to be integrated with the embrittlement data.

Status: A computer data base containing data from 81 surveillance reports representing 53 reactors and 66 capsules is available and a surveillance manual is to be available by March 1981.

R. A. Wullaert, P. McConnel and W. Oldfield, "Status of Radiation Embrittlement Trend Curve Data Base Re-evaluation," Proceedings of the Third ASTM-Euratom Meeting on Reactor Dosimetry, Ispra, Italy, September 1979, pp. __.

K. E. Stahlkopf, G . R. Odette and T. U. Marston, "Radiation Damage Saturation in Reactor Pressure Vessel Steel: Data and Preliminary Model," Proceedings, Fourth International Conference on Pressure Vessel Technology, London, May 1980, pp. _ .

RP1553-1 Evaluation of Irradiation Response of Reactor Pressure Vessel Materials

Prime Contractor: Combustion Engineering, Inc. (J. J. Koziol)

Duration: July 19, 1979 to May 18, 1981

Project Cost: $170,616

EPRI Project Manager: T. U. Marston

Objective: The objective is to provide a means for predicting the Charpy V-notch toughness reponse of reactor pressure vessel materials to power reactor neutron exposures by taking into account material and operating variables not presently considered but suspected of contributing to the observed variability in current predictive procedures. This effort will utilize existing commercial power reactor pressure vessel Charpy surveillance data and all pertinent information associated with material fabrication and irradiation conditions available from these programs.

Strategy: Tasks I and II will compare the indices obtained by using various computer curve fitting techniques with reported indices obtained by manual curve fitting. These comparisons will be conducted on pre- and post-irradiation data contained in the EPRI data bank. The proposed curve fitting techniques will be statistically evaluated with respect to their ability to produce curves which accurately represent the classical regions of Charpy transition curves and selected indices. Task III will statistically examine Charpy transition temperature shift and decreases in upper shelf energy, using analytical models coupled with actual test data, in an attempt to establish predictive methods for these parameters.

Status: A review and update of the EPRI surveillance data base was made in 1980. Statistical analysis of the data is progressing after final completion of Tasks I and II.

J. D. Varsik, S. T. Byrne, "An Empirical Evaluation of the Irradiation Sensitivity of Reactor Pressure Vessel Materials," Effects of Radiation on Structural Materials, ASTM STP 683, 1979, pp. 252-266.

RP1553-2 Application of ALN Modeling to Radiation Embrittlement

Prime Contractor: Adaptronics, Inc.

Duration: June 25, 1979 to March 31, 1981

Project Cost: $71,962

EPRI Project Manager: T. U. Marston

Objective: The objective is to establish the foundation for developing a statistically-based model for the dependence of toughness-test response on temperature and fluence, incorporating the irradiation time, impurity chemistry, neutron energy density, material type and mechanical properties for base, weld and heat-affected-zone nuclear reactor steels. A successful radiation embrittlement toughness response model will contribute significantly to the development of statistically-based trend curves.

Strategy: Univariate statistical analysis and cluster/pattern recognition methods will be used to characterize the EPRI irradiated data base prior to modeling. The Adaptive Learning Network (ALN) modeling concept will be used to develop the required statistically-based model.

Status: The EPRI data base on irradiated steels has been received and installed on the contractor's computer. Complete empirical analyses of the raw data are available. Analysis of the curve fitted data is in process.

RP886-1 Analysis of Radiation Embrittlement Reference Toughness Curves

Prime Contractor: Fracture Control Corporation (R. A. Wullaert)
Duration: September 11, 1976 to March 31, 1980
Project Cost: $642,076
EPRI Project Manager: T. U. Marston

Objective: The objective of this program is to develop a statistically valid radiation embrittlement data base for use in the critical evaluation of the procedures for predicting the fracture toughness of irradiated reactor pressure vessel steels as currently specified in NRC Regulatory Guide 1.99

Strategy: Task I involves the development of statistically-based reference toughness curves for irradiated materials. Task II is directed towards establishing the relationship between radiation embrittlement measured by the Charpy V-notch test and several fracture mechanics tests. Task III is concerned with the acquisition, qualification, improvement and ultimate application of radiation exposure parameters and the modeling of radiation damage.

Status: All tasks have been completed and a final report has been prepared. A statistically-based reference toughness procedure has been submitted to ASME. Several advanced concepts have been developed for measuring radiation embrittlement from Charpy surveillance specimens. The uncertainties associated with radiation analysis have been defined and a radiation damage model has been developed.

W. L. Server and W. Oldfield, "Nuclear Pressure Vessel Steel Data Base," EPRI NP-933, December 1978.

R. A. Wullaert and W. L. Server, "Fracture Toughness of Nuclear Pressure Vessel Steels from Small Specimens," Transactions of the 5th International Conference on Structural Mechanics in Reactor Technology, Vol. G, paper G 2/1.

G. R. Odette, "A Quantitative Analysis of the Implications of the Accuracy of Dosimetry to Embrittlement Predictions: Past, Present and Future," Proceedings Third ASTM-Euratom Meeting on Reactor Dosimetry, Ispra, Italy, September 1979.

W. L. Server, et al., "Analysis of Radiation Embrittlement Reference Toughness Curves," Final report EPRI NP-1661, Palo Alto, California, January 1981.

RP886-2 Evaluation and Prediction of Neutron Embrittlement in Reactor Pressure Vessel Materials

Prime Contractor: Naval Research Laboratory (J. R. Hawthorne)

Duration: January 10, 1977 to September 15, 1981

Project Cost: $1,150,000

EPRI Project Manager: T. U. Marston

Objective: Objectives are to develop a data base for the evaluation of current radiation embrittlement project methods and for the development of improved procedures, to investigate the relationship between radiation effects measured by the Charpy-V (C_v) test method and fracture mechanics tests methods, to determine the radiation embrittlement sensitivities of a broad range of reactor vessel materials, and to experimentally assess the effects of selected composition variations.

Strategy: Changes in the fracture resistance of representative reactor vessel materials are determined and compared with irradiation at 288°C in a nuclear test reactor. Three neutron fluence levels are employed. Radiation sensitivity is judged from the degradation of C_v notch ductility and the degradation of fracture toughness determined by fatigue precracked C_v and compact toughness (CT) tests. Results test embrittlement projection methods for a wide range of composition and metallurgical variations.

Status: The irradiations are complete. All of the impact tests are complete and approximately 80% of the compact fracture specimens are tested. Findings to date indicate that current embrittlement projection methods are inconsistent, accurate in some cases to highly conservative in others. Numerous examples of J-resistance curves for irradiated materials are presented.

J. R. Hawthorne, Editor, "The NRL-EPRI Research Program (RP886-2), Evaluation and Prediction of Neutron Embrittlement in Reactor Pressure Vessel Materials, Annual Progress Report for CY 1979 Part I Dynamic C_v, PCC_v Investigations," NRL Report 4431, December 31, 1980.

RP1326-1 Development of a Crack Arrest Toughness Data Bank for Irradiated RPV Materials

Prime Contractor: Westinghouse Electric Corporation (T. R. Mager)

Duration: February 16, 1979 to April 30, 1982

Project Cost: $1,050,843

EPRI Project Manager: S. W. Tagart, Jr.

Objective: The overall objective is to determine the effects of fast neutron exposure on the crack arrest toughness of reactor pressure vessel materials.

Strategy: Specimens fabricated from four materials characteristic of current reactor vessel materials will be irradiated in the University of Virginia Research Reactor to a fluence of about 1×10^{19} n/cm^2 (E > 1.0 MeV). The irradiated specimens will be tested at Battelle Columbus Laboratories and Westinghouse Research Laboratories. Test data will be evaluated and compared to test data for unirradiated materials of the same types. Statistical analyses will be performed to determine the effect of irradiation on crack arrest toughness.

Status: Due to extension of the irradiation schedule at the University of Virginia reactor, this project is being extended over a longer time period. The first irradiation cycle has been completed and testing is in progress. The second cycle is currently in progress and the third cycle will begin in April 1981. Two semiannual progress reports have been received and reviewed by EPRI.

G. T. Hahn, et al., "Critical Experiments, Measurements, and Analyses to Establish a Crack Arrest Methodology for Nuclear Pressure Vessel Steels," BMI-2025, 1979.

T. R. Mager, "Development of a Crack Arrest Toughness Data Bank for Irradiated RPV Materials, semiannual technical progress reports 1 and 2 through March 1980.

RP1021-1 Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel

Prime Contractor: Westinghouse Electric Corporation (T. R. Mager)

Duration: June 20, 1977 to May 31, 1981

Project Cost: $1,370,614

EPRI Project Manager: T. U. Marston

Objective: The overall objective is the development of an optimal in-situ thermal annealing methodology for reactor vessels which maximizes fracture toughness recovery, minimizes reexposure sensitivity, and minimizes downtime.

Strategy: Through irradiation, annealing, and testing of specimens fabricated from material typical of reactor pressure vessel weldments, and analysis and evaluation of the resulting data, determine fracture toughness recovery as a function of annealing time and temperature, determine post-anneal embrittlement sensitivity, and the kinetics of embrittlement, thermal annealing, post-anneal embrittlement, and subsequent reannealing for the materials. Based on the results, establish optimal thermal annealing procedures for field application.

Status: Irradiation of the three sets of capsules at the University of Virginia Research Reactor has been completed. Testing of irradiated specimens and analysis of test results have been nearly completed for all irradiations. System evaluation for high temperature (dry) anneal are underway.

J. S. Schlonski, "Feasibility of Operating the Reactor Coolant System at Design Temperature for Reactor Vessel Annealing," EPRI RP1021-1 Topical Report, 1978.

S. L. Anderson, "Characterizationn of the University of Virginia Research Reactor Radiation Environment," EPRI RP1021-1 Topical Report, 1979.

T. A. Meyer, "Design and Fabrication of Specimen Irradiation Capsules for the Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel Program," EPRI RP1021-1 Topical Report, 1979.

T. R. Mager, et al., "Feasibility of and Methodology for Thermal Annealing an Embrittled Reactor Vessel" Semiannual Technical Progress Reports No. 1-6.

RP1021-2 Feasibility and Methodology for Thermal Annealing an Embrittled Reactor Vessel

Prime Contractor: University of Virginia (B. Shriver)

Duration: March 1, 1978 to June 30, 1982

Project Cost: $584,103

EPRI Project Manager: T. U. Marston

Objective: The primary objectives of this project are to determine the feasibility of, and develop the methodology for thermal annealing an embrittled reactor pressure vessel. This annealing treatment is designed to restore the mechanical properties reduced by neutron irradiation.

Strategy: The strategy of the project is as follows: determine the extent of fracture toughness for radiation sensitive materials as a function of thermal treatment; determine the post anneal embrittlement, anneal, reembrittlement and reanneal; if significant residual recovery is possible, establish optimal thermal anneal procedures for field application and evaluate technical risks of the proposed anneal procedures.

Status: The irradiations are being conducted under this contract at the University of Virginia test reactor for contracts RP1021-1, RP1326-1 and RP1325-4.

Various UVAR semiannual reports prepared by Professor B. Shriver concerning reactor operation; no specimen irradiation.