LETTER REPORT ON REACTOR VESSEL

BRITTLE FRACTURE CONCERNS

IN B&W OPERATING PLANTS

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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 injected into 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 $100^{o}F$ subcooled conditions at the core outlet is maximized, resulting in increased thermal stress during the transient.
5. A worst-case HPI fluid temperature of $40^{o}F$ 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. Bounding Assumptions

   Analyses were performed assuming no heatup of HPI due to any o 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. Mix Assumptions

   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 ±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

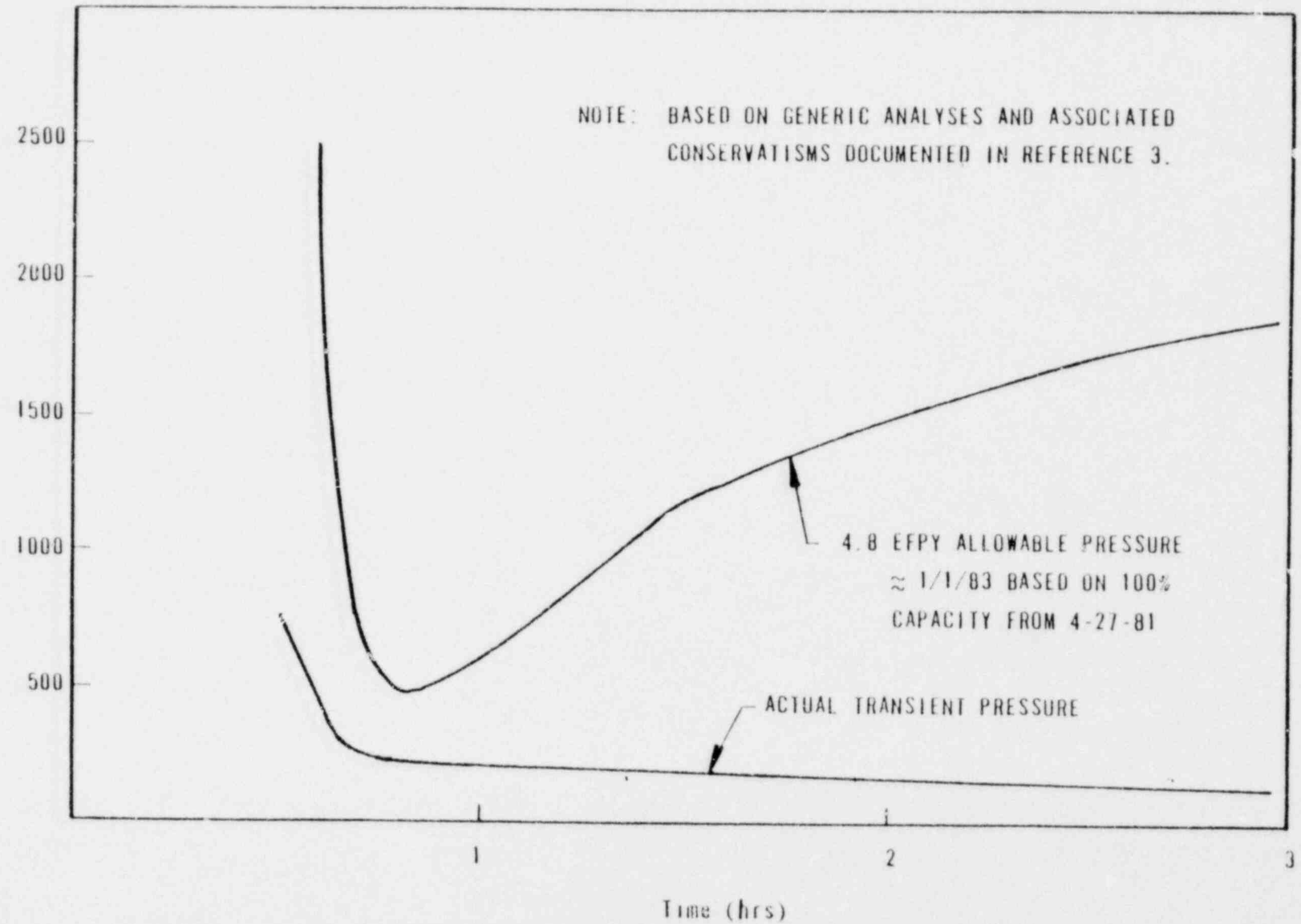
Primary System Response During Overcooling Transients

Case	Description	Cooldown Rate	Minimum Downcomer Temperature	Comment
1	BAW 1648 Bounding Analysis(3)	460F in 60 Seconds ($460^{o}F$/min)	90F	No temperature recovery No repressurization* (90F BWST)
2	BAW 1648 Mix Analysis(3)	445F in 40 minutes ($11.1^{o}F$/min)	90F	No temperature recovery No repressurization* (40F BWST)
3	Unmitigated Large Steam Line Rupture(7)	320F in 10 Minutes ($32^{o}F$/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



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(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 1 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.