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AUXILIARY FEEDWATER SYSTEMS RELIABILITY ANALYSES

A Generic Report for Plants With Babcock & Wilcox Reactors

by

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EXECUTIVE SUMMARY

This report presents a generic summary of the analysis methods and results of a reliability study of Auxiliary Feedwater Systems (AFWS) at operating plants with Babcock & Wilcox designed Nuclear Steam Supply Systems.

The objectives of this report were:

1) To identify, through reliability based insights, dominant contributors to AFWS unreliability.

2) To assess the relative reliability of B&W operating plant Auxiliary Feedwater Systems.

Dominant contributors to unreliability are identified in Table 2. These contributors vary widely in significance, ranging from the relatively unavoidable contribution of preventive maintenance to AC dependencies which preclude system operation on loss of AC power. In every case where significant contributors were identified, improvements by design and/or procedural changes should be achievable. These contributors provide a rational basis for design changes to improve AFWS reliability.

A comparative perspective on the range of reliabilities which can be expected from B&W operating plant Auxiliary Feedwater Systems is shown in Figure 1. The relationship of these values to the NRC-calculated reliabilities for plants of Westinghouse and Combustion Engineering design is not straight forward in that certain assumptions appear to be more conservative in the B&W analyses than in the NRC analyses; the basis for this belief is explained in Appendix B.

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1.0 INTRODUCTION

This report presents a generic summary of the analysis methods and results of a reliability study of Auxiliary Feedwater Systems at operating plants with Babcock & Wilcox (B&W) designed Nuclear Steam Supply Systems.

The Auxiliary Feedwater System functions as an emergency system for the removal of heat from the primary system when main feedwater is not available. Some B&W operating plants refer to this system as an Emergency Feedwater System; however, througnout this report, the term Auxiliary Feedwater System (AFWS) will be used.

Also contained in this report is an overview of AFWS designs at the B&W operating plants, a description of assumptions used during this study and appropriate limitations which should be observed when considering the results of the study.

1.1 Background

As one outgrowth of the incident at Three Mile Island-2, the NRC requested all operating plants to consider means for upgrading the reliability of their Auxiliary Feedwater Systems. As a part of the response to this request, the B&W Owners Group utilities asked B&W to perform reliability analyses of the existing Auxiliary Feedwater Systems at each B&W operating plant. The ultimate objective of this work is to determine what changes, if any, will improve AFWS reliability.

The NRC has conducted similar analyses for Westingnouse and Combustion Engineering plants; descriptions of those analyses and the results are in References 1 and 2. The NRC requested that the B&W analyses be performed within a time frame and on a basis consistent with the NRC's own analyses. Accordingly, the scope of B&W's study and arrangement of the schedule were made in agreement with the NRC's request.

B&W performed the requested analyses and has issued to each of the utilities a report containing a plant specific AFWS reliability evaluation. A generic summary of the analysis methods and results contained in these plant specific reports are presented herein.

1.2 Objectives

The objectives of this study were:

- To perform simplified analyses to assess the relative reliability of B&W operating plant Auxiliary Feedwater Systems. It was intended that these analyses would be performed on a basis consistent with that used by the NRC in analyses for Westinghouse and Combustion Engineering plants. It was further intended that such consistency would be achieved by use of the same evaluative technique, event scenarios, assumptions and reliability data used by the NRC.
- To identify, through the development of reliability-based insight, dominant contributors to AFWS unreliability.

1.3 Scope

Auxiliary Feedwater Systems at the following B&W operating plants were analyzed:

Rancho Seco
Oconee Units I, II & III
Crystal River-3
Davis-Besse-1
Arkansas Nuclear One-1
Three Mile Island-1

The analysis for each plant was based on the configuration of the Auxiliary Feedwater System as it existed on August 1, 1979, but also included were any near-term changes which were already in process and which would be in place by December 3, 1979. An exception was made for the Three Mile Island-1 plant; a configuration date of early 1980, corresponding to the earliest anticipated startup of this plant was used.

Three event scenarios were considered in this study:

- Case 1 - Loss of Main Feedwater with Reactor Trip (LMFW)
- Case 2 - LMFW coincident with Loss of Offsite Power (LMFW/LOOP)
- Case 3 - LMFW coincident with Loss of all AC Power (LMFW/LOAC).

These event scenarios were taken as given; that is, postulated causes for these scenarios and the associated probabilities of their occurrences were not considered. Additionally, external common mode events (earthquakes, fires, etc.) and their effects were excluded from consideration.

For each of the three cases, system reliability as a function of time was evaluated. Three times were considered: 5, 15 and 30 minutes following LMFW (Refer to Section 2.2). A total of 54 detailed fault tree analyses were performed covering the six AFWS designs with three event scenarios and at three times for each event. Each plant's specific event tree can be found in the respective plant specific report (References 4-9).

1.4 Summary and Conclusions

The principal result of this study is the identification of dominant contributors to AFWS unavailability for each plant. Pending further evaluation by the utilities, these contributors may provide a rational basis for the selection of design changes to improve AFWS reliability.

The dominant contributors identified in Table 2 vary widely in significance, ranging from the relatively unavoidable contribution of preventive maintenance, to AC dependencies which will preclude system operation on loss of AC power. In every case where significant contributors were identified, improvements by design and/or procedural changes should be achievable. If appropriate modifications are accomplished, B&W operating plant AFW Systems will exhibit, as a group, reliabilities close to the maximum reliability attainable for real, two-train systems.

The quantitative results of these analyses, shown in Figure 1, provide a general comparative perspective on the range of reliabilities which can be expected from B&W operating plant Auxiliary Feedwater Systems. Although it was intended that this study closely match the NRC study for Westinghouse and Combustion Engineering Auxiliary Feedwater Systems, the results of the two studies should not be directly compared; see Appendix B.

1.5 Limitations

Careful consideration must be given to the validity and applicability of the results of this study, these results could be misleading if taken out of context. Appropriate limitations on the use of these results include:

(1) Relative reliability standings. This report presents (Figure 1) the relative reliability standings of all the B&W plants, and while these results can show major differences, small differences between plants are not significant. Further, no direct comparison of the quantitative results for the B&W plants to the NRC calculated results for Westinghouse and C-E plants should be made without a thorough understanding of the analyses. Even though a concerted effort was made to maintain uniformity with analysis methods and assumptions used by the NRC, B&W believes that certain inconsistencies exist. (See Appendix B.)

(2) Absolute values of availability. This analysis resulted in only relative reliabilities and not absolute values of AFWS unavailability. Any inference of realistic AFWS reliability must address the probability of occurrence of the three event scenarios in addition to considering other defects which may accompany the conditions producing these scenarios.

(3) Dominant failure contributors. This analysis identified the dominant contributors to system unavailability; however, this report did not explore possible modifications to those contributors. While in some cases a simple change appears feasible, other cases are obviously complex situations with many possible solutions. Each utility must decide if cost-effective modifications are available for their dominant contributors. (Dominant contributors are discussed in Section 4.2.)

2.0 DESCRIPTION OF ANALYSIS

2.1 Analysis Method

The analysis method used to evaluate the reliability of Auxiliary Feedwater Systems in operating B&W plants involved the construction and analysis of fault trees. The techniques used in this effort were consistent with those described in the Reactor Safety Study, WASH-1400 (Reference 3).

The result of this analysis is the point unavailability of the AFWS, under three scenario conditions and at three points in time following the initial existence of conditions requiring AFWS initiation. Point unavailability is equivalent to the probability that the system will be unavailable at the point in time at which a demand is placed on it.

To support this analysis, each utility with a B&W NSSS furnished to B&W the plant specific system drawings, electrical schematic diagrams, operating, test and maintenance procedures and technical specifications for the Auxiliary Feedwater System and pertinent support systems. From this systems data, B&W extracted information necessary to prepare a detailed AFW system description (References 4 thru 9). This description was reviewed for accuracy by the utility to ensure that the system analyzed was, indeed, the system that physically exists at the site.

A fault tree was constructed for each utility based on this detailed system description. The top level event in the fault tree was failure to achieve mission success (defined in Section 2.2). Top level sub-branches of the tree generally involved multiple failures resulting in the unavailability of all feedwater trains and included unavailability arising from preventive maintenance activities. Examples of multiple failures leading to system unavailability of a two-train system include: failure of the pumps in both trains; or combination failures such as failure of one pump coupled with a discharge path failure in the opposite train and no available discharge cross-tie.

From the top level event, fault tree branches were expanded downward to a level of detail corresponding to unavailability data which was supplied by the NRC. This level of detail was typically that associated with component failure cause (valve plugging, pump control circuit failure, etc.)

The NRC-supplied unavailability data consisted of expected unavailability numbers for typical fluid and control system hardware, human failure probabilities as a function of time, and unavailability associated with preventive maintenance. This data was obtained as a part of Reference 1, and is shown in Appendix A. The data was supplemented when necessary by direct consultation with the NRC staff and by engineering judgment. (The NRC has emphasized that these input data are largely unverified estimates of human and component reliability. According to the NRC, errors as large as an order of magnitude up or down may exist in this data. In spite of this uncertainty, such data can provide a uniform basis for obtaining reliability results for plants with substantially different system designs. Because of this uncertainty, absolute values of calculated reliability must be strongly de-emphasized, and even relative reliability standings are subject to uncertainty.)

After construction of the fault tree, unavailability analyses were performed. These analyses were accomplished by inserting the NRC-supplied data at the bottom-level basic events of the fault tree and then working upward with hand calculations to assess the cumulation of unavailability. Each tree was analyzed a total of nine times; this was necessary to incorporate appropriate modifications for the three event scenarios at each of three times following the initial demand.

Performing the analyses, at the level of detail described above, provided insights into the relative importance of various contributors to overall system reliability. Thus, the analysis approach used permitted the identification of major failure contributors which was a major objective of the study.

2.2 General Assumptions and Criteria

Agreement was reached with the NRC staff regarding the assumptions and criteria used in this study, with the goal of obtaining results which were on a consistent basis with those produced by the NRC in its Westinghouse and Combustion Engineering analyses. The assumptions and criteria which were used in this study and which have general applicability are described below. Other, plant specific, assumptions were used and these are contained in the reliability reports for each utility (References 4-9).

1) Definition of Mission Success - In order to evaluate the contribution of system components to overall reliability, it was necessary to determine to what extent failure of those components might prevent successful accomplishment of the AFWS mission. This in turn requires an explicit definition of mission success. The definition adopted for this study was the attainment of flow from at least one full capacity pump (or from at least two half-capacity pumps) to at least one steam generator. Attainment of flow from only one half-capacity pump was not considered system success.

   System reliability was calculated at times of 5, 15, and 30 minutes following the existence of initiating conditions to allow for a range of operator action. These times were specifically chosen because NRC-supplied operator reliability data for these times was available; these times are reasonable and consistent with LMFW mitigation for B&W plants. In their study, the NRC staff has used steam generator dryout time as a criterion for successful AFWS initiation, and the 5-minute case represents a comparable result for B&W plants with anticipatory reactor trips on LMFW. However, steam generator dryout itself does not imply serious consequences; a more appropriate criteria is the maintenance of adequate core cooling. Recent ECCS analyses (Reference 10) have shown that adequate core cooling can be maintained for times in excess of 20 minutes without AFWS operation, providing that at least one High Pressure Injection Pump is operated. (For Davis-Besse-1, the requirements are contained in References 7 and 11.)

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In general, the loss of flow, resulting from random component failures after successful AFWS initiation, was not considered within the scope of this study. However, system characteristics or component limitations which were <u>known</u> to potentially restrict the duration of system operation (to less than 2 hours) were considered in accordance with NRC guidance. Such limitations were included by assuming that they resulted in instantaneous unavailability of the affected components unless the underlying causes were correctable within 5, 15 or 30 minutes. It must be emphasized that this method for accounting for latent failures results in a very conservative analysis. It may not take credit for successful AFWS operation until failure, nor does it allow for the possibility that corrective or mitigating measures can be used (such as restoring power or cycling components on and off).

2) <u>Power Availability</u> - The following assumptions were made regarding power availability:

   LMFW - All AC and DC power was assumed available with a probability of 1.0.

   LMFW/LOOP - All DC power was assumed available with a probability of 1.0. Where applicable, one diesel generator was assumed available with a probability of 1.0 and the other was assumed unavailable with a probability of 10^{-2}.

   LMFW/LOAC - DC and battery-backed AC were assumed available with a probability of 1.0.

3) <u>Interconnections with Other Units</u> - In general, no credit was taken nor any penalty assigned for steam, electric power or auxiliary feedwater supplied from, or diverted to, other adjacent plants.

4) <u>NRC-Supplied Data</u> - NRC-supplied unreliability data for hardware, operator actions and preventive maintenance were assumed valid and directly applicable.

5) <u>Coupled Manual Actions</u> - Manual initiation of valves with identical function and the same physical location was considered coupled. Such valves were assumed to be both opened manually or both not opened. The case in which one valve was opened and the other valve was left closed was not considered.

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6) Degraded Failures - This was a binary type analysis as defined in Reference 3. Degraded failures were not considered; that is, components were assumed to operate properly or were treated as failed.

7) Small Lines Ignored - Typically, lines on the order of 1-inch were ignored as possible flow diversion paths.

8) Steam Supply for AFWS Turbines - Adequate steam to the turbine-driven-pump turbines was assumed for the 15 and 30 minute cases. These turbines and pumps are designed to deliver water to the steam generators using steam remaining in the steam lines after generator dryout.

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3.0 OVERVIEW OF B&W AUXILIARY FEEDWATER SYSTEMS

A summary description of the major characteristics of Auxiliary Feedwater Systems at B&W operating plants is contained in Table 1. This information was extracted from plant specific reliability reports which were prepared for each utility (References 4-9). As indicated in the table, there are many functional similarities between the AFWS analyzed. These similarities and some exceptions are summarized below.

All AFWS are capable of providing auxiliary feedwater to one or both steam generators under automatic (or manual) initiation and control.

Each system consists of multiple feedwater trains with a combined capacity of twice the flow of a nominal full capacity pump. This capacity is achieved by the use of at least one full-capacity turbine-driven pump and, with the exception of Davis-Besse-1, which has two turbine-driven pumps, each has either one full-capacity or two half-capacity motor-driven pumps. With the exception of Crystal River-3 and the Oconee Units, all AFW turbines, motors and pumps are self-sufficient entities without dependence on secondary support systems.

Each AFWS has multiple suction sources available, including the condenser hotwell or other backup water supply. Switchover to the backup water supply requires manual action except for Davis-Besse-1 for which this action is automatic.

Motive power for the motor-driven pump(s) is obtained from one (or two, as applicable) nuclear service busses. These busses are backed by diesel generators or, at Oconee, hydro generators. Manual loading of the pump motors onto the diesel generators is required at Rancho Seco and Crystal River-3. In each system, steam for the AFWS turbine(s) may be obtained from either steam generator.

Conditions which will cause AFWS initiation vary between plants with the only common initiating condition being loss of both main feedwater pumps. Every system will be initiated by at least one other condition; examples include: loss of all four reactor coolant pumps or low steam generator level. All AFWS pump initiation circuitry is battery-backed and, except for Arkansas Nuclear One-1, is independent of the Integrated Control System (ICS).

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All AFWS but Davis-Besse-1 and the Oconee Units control the flow of auxiliary feedwater to the steam generators by flow control valves under ICS control. Oconee uses separate steam generator level control circuits and Davis-Besse-1 controls steam generator level by varying turbine speed.

With correct system alignment and no component failures, none of the plants require manual action to achieve mission success for Case 1 (LMFW). In Case 2 (LMFW/LOOP), none of the plants except the Oconee Units require manual action to obtain flow from the turbine-driven pump(s), but manual actions described earlier are required to energize the motor-driven pumps at Rancho Seco and Crystal River-3. In Case 3 (LMFW/LOAC), only Rancho Seco and Three Mile Island-1 will achieve sustained auxiliary feedwater flow from the turbine-driven pump without manual actions.

4.0 RELIABILITY EVALUATION

4.1 Quantitative Analysis Results

The quantitative results of the fault tree analyses are presented in Figures 1A, B and C. Indicated in these figures are the Auxiliary Feedwater System unavailabilities for each B&W operating plant for each of the three scenario cases and at each time 5, 15 and 30 minutes. These figures provide a general comparative perspective on the range of reliabilities which can be expected from B&W operating plant Auxiliary Feedwater Systems. Limitations described in Section 1.5, should be observed when considering data presented in these figures.

Shown in each figure is an approximate upper limit for the reliability of a two-train AFW system in which the pump in one train is electric-powered from a diesel generator during loss of offsite power. This limit is calculated for a two train system in which each train consists of one pump with drive, one check valve and one normally open flow control valve. Pump discharges are interconnected with a crosstie and pump suctions are connected to a "perfect" source. The system has no common mode vulnerabilities or human dependencies. This upper limit, which does not apply to Davis-Besse 1 in Cases 2 and 3 because of their two-turbine system, represents the reliability of an idealized system using only the number of components needed to approximate optimum reliability; this limit is calculated from NRC-supplied component failure data. The minimum reliability in each case represents unavailability of the system (i.e., probability of unavailability is 1.0). The presentation of reliability results in the format of Figure 1 demonstrates the range of reliabilities against a frame of reference which has physically meaningful limits for each case.

Consistent with the results reported by the NRC for Westinghouse and Combustion Engineering Plants (References 1 and 2), B&W operating plant AFWS designs exhibit more than an order of magnitude variability in the calculated reliability for each of the three event scenarios considered.

The effect of degraded power availability is indicated clearly by the differences in the results for each of the three cases. Except for the Oconee Units, the loss of offsite power results in a relatively small decrease in system availability (typically one order of magnitude or less), primarily resulting from the assumed unavailability of one of the two diesel generators (with a probability of 10^{-2}). However, as indicated by the Case 3 results, a loss of all AC power will have significant consequences for all units. In Case 3, all but two of the units have AC dependencies which would inhibit system operability.

The effect of corrective operator actions is also shown in Figure 1. As the time allowed for operator action increases from 5 to 15 and 30 minutes, system unavailability usually improves because human reliability improves and because the range of possible operator action increases (to include for example, manual actions outside the control room). Reflecting the NRC-supplied human reliability data, this improvement is much more pronounced in the interval between 5 and 15 minutes than in the interval between 15 and 30 minutes. This improvement is also somewhat more pronounced in Case 1 than in Cases 2 and 3 where degraded power availability tends to reduce the number of available options for operator action.

In atypical cases, system reliability may decrease with time, even allowing for increased probability for operator corrective actions. This results from the treatment of latent failures discussed in Section 2.2.

4.2 Dominant Failure Contributors

A summary tabulation of dominant failure contributors revealed during the fault tree analyses is presented in Table 2. It appears that improvement of AFWS reliability, based on modifications of hardware-related failure contributors, should be achievable for all B&W plants. In no case are the contributors so extensive in nature that the inherent AFWS design is unacceptable. Improvement in AFWS reliability with the removal of dominant contributors is expected to be dramatic in some cases. For example, the addition of a valve position indicator may result in a calculated system reliability improvement of nearly an order of magnitude.

The most common dominant contributor for Case 1 is outage for preventive maintenance-related activities. Such outages reduce system redundancy and increase the likelihood of unavailability if AFWS use is required. Other typical contributors affecting more than one plant include: flow diversion through normally-closed manually-operated recirculation test valves which may be left open inadvertently, and failure to obtain pump initiation and/or control valve opening because both AFWS trains rely on common initiation/control circuit components.

In general, the loss of offsite power does not impose significant new conditions on the AFWS such that new and substantially different failure contributors become dominant. Thus, Case 2 major failure contributors tend to be identical with those identified during the Case 1 analyses. Specific exceptions to this rule include: human failures associated with the manual loading of the motor-driven pumps onto diesel generator-backed busses at Rancho Seco and Crystal River-3; and human failure to perform actions necessitated by automatic load shedding at Oconee.

With the exception of Three Mile Island-1 and Rancho Seco, the Case 3 analyses indicate significant AC dependencies for Auxiliary Feedwater Systems. These dependencies may be direct as is the case for Davis-Besse-1 and Arkansas Nuclear One-1 where certain valves required for AFWS mission success are AC powered; or the dependencies may be indirect, as is the case for Crystal River-3 and the Oconee Units, where AFWS support systems require AC power for continued AFWS operation.

The significance of failure contributors must be carefully evaluated before design and/or procedural changes are recommended. Such evaluation is required because even the significance for the same contributor varies widely between plants. Such variation exists because the importance of failure contributors is distributed differently for different AFWS designs. A dominant failure contributor for a plant like Davis-Besse-1, which has a relatively uniform distribution of potential failure importance, may be almost insignificant by comparison to a dominant contributor for a plant with salient failure contributors. It is necessary to consider such factors in order to determine the most effective utilization of resources for reliability improvement.

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4.3 Single Point Vulnerabilities

A review of Table 2 reveals that two of the AFWS designs (Davis-Besse and Oconee) do not have single point vulnerabilities in Case 1. In Case 2 only one AFWS (Davis-Besse) has no single point vulnerabilities. In Case 3, all plants have single point vulnerabilities.

TABLE 1. SUMMARY OF MAJOR CHARACTERISTICS OF B&W OPERATING PLANT AFW SYSTEMS

	Rancho Seco	Oconee-I,II,III	Crystal River-3	Davis-Besse-1	Arkansas Nucl. One-1	Three Mile Island-1
Pumps	1 turbine/motor driven 1 motor driven	1 turbine driven 2 ½cap. motor driven	1 turbine driven 1 motor driven	2 turbine driven	1 turbine driven 1 motor driven	1 turbine driven 2 ½cap. motor driven
Primary Suction Source	250,000 g. CST	50,000 g. USTA+B for TDP UST+100,000 g. Cond. Hotw. for MDP	150,000 g. CST	2 CST's each 250,000 g.	107,000 g. CST	2 CST's each 150,000 g.
Alter. Suction Source	Canal & reservoir connector	Condensor Hotwell	Condensor Hotwell	2 Svc. Water Trains	Nucl. Svc. Water Sys.	Riv. Water Sys.
Switchover to Alt. Suction	Manual	Manual for TDP	Manual	Auto.	Manual	Manual
Discharge Crosstie	Yes, with N.O. valves	No (N.C. paths not considered) Each MDP feeds 1 S/G, TDP feeds both	Yes, two with check valves	Yes with N.C. valves SFRCS/man. control	Yes with N.O. valves	Yes any pump feeds any S/G
Backup Power	2 diesel gen.	Keowee hydro gen.	2 diesel gen.	2 diesel gen.	2 diesel gen.	2 diesel gen.
Common Steam Supply Header Fed From both S/G	Yes	Yes	Yes	No, separate stm. supply lines with cross-over connections under SFRCS control	Yes	Yes
Pump Initiation — TDP	ESFAS, 4 RCP trip, 2 MFWP trip	2 MFWP Lo Disch Press 2 MFWP Trip	2 MFWP trip 2 S/G Lo level	1 MFW Vlv. Hi Rev. ΔP 1 S/G Lo Lvl., 4 RCP Trip, S/G Lo P*	2 MFWP Trip, 1 s/g Lo Lvl. 4 RCP Trip	2 MFWP Lo ΔP, 2 MFWP Trip 4 RCP Trip
MDP	Same minus ESFAS	Same	Same	N/A	Same	Same minus 2 MFWP Trip
Location	Ext. to ICS	Ext. to ICS	Ext. to ICS	SFRCS	All within ICS	Ext. to ICS
AFW Control & Valves	ICS Contr. for Flow Control Vlvs. S/P's for Loss of 4 RCP, 2 MFWP	S/G Lvl. Contr. Ckts. for each S/G flow contr. vlvs	ICS contr. for flow contr. vlvs.	Turbine speed contr, speed-contr. vlvs, SFRCS Isol. vlvs. All contr. sep. from ICS	ICS contr. for flow contr. vlvs. S/P's for Loss of 4 RCP, 2 MFWP	ICS contr. for flow contr. valves. S/P's for Loss of 4 RCP, 2 MFWP
Operator Actions For Sustained AFW Flow — Case 1	None R'qd.	None R'qd.	None R'qd.	None R'qd.	None R'qd.	None R'qd. (Open 6" Stm Supply)
Case 2	Man. Load of MDP on D.G. (if TDP fails)	Open TDP Cool. Water Vlv., restore load shed PWR	Man. Load of MDP (if TDP fails)	None R'qd.	None R'qd.	None R'qd. (Open 6" Stm. Supply)
Case 3	None R'qd.	None Avail.	None Avail.	Man. open. AC Vlvs.	Man. open AC Vlvs.	None R'qd. (Open 6" Stm. Supply)

Note: For details, refer to plant specific draft reports (References 4-9)

TDP - Turbine Driven Pump
MDP - Motor Driven Pump
CST - Condensate Storage Tank
UST - Upper Surge Tank
RCP - Reactor Coolant Pump
MFWP - Main Feedwater Pump
S/G - Steam Generator
S/P - Set Point
ICS - Integrated Control System

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TABLE 2 - MAJOR FAILURE CONTRIBUTORS

	Rancho Seco	Oconee-I,II,III	Crystal River-3	Davis Besse-1	Arkansas Nucl. One-1	Three Mile Island-1
Case 1: LMFW	1) Flow diversion from both trains via recirc. valve FWS-055, if inadvertently open. 2) Outages for preventive maintenance.	1) Turbine support system failures eg. aux. lube oil pump. 2) Turbine pump bearing failure if valve LPSW-137 doesn't open. 3) Loss of suction because upper surge tanks are not replenished. 4) Flow diversion via recirc. valve FDW-88, if inadvertently open.	1) Valve plugging in a common cooling water line to both pumps. 2) Outages for preventive maintenance.	1) Preventive maintenance of one train coupled with random failures in the other can defeat mission success.	1) Preventive maintenance outages. 2) Failure to obtain system initiation because failure of common components in the initiation and control equipment for both trains. 3) Flow diversion via FW11A,12A,11B or 12B. 4) Suction related failures (incorrect alignment of CV2803 and CV2800).	1) Failure to obtain feedwater flow because of actuation circuit failures common to both trains. 2) Preventive maintenance outages. 3) Isolation valves inadvertently left closed after pump testing.
Case 2: LMFW/LOOP	Case 1 Contributors plus: 1) Failure to manually load motor driven pump onto diesel.	Case 1 Contributors plus: 1) Loss of cooling water to turbine pump because LPSW-137 is load shed. 2) Loss of suction for turbine unless C-391 is opened and manual loading of hotwell pumps on 4160 VAC busses. 3) MS-92 staying open because MS-87 or MS-129 have failed open on loss of air-inadequate steam for turbine.	Case 1 Contributors plus: 1) Failure to manually load the motor driven pump onto the diesel.	Case 1 Contributor	Case 1 Contributors	Case 1 Contributors
Case 3: LMFW/LOAC	Case 1 Contributors involving turbine driven pump.	Case 1 and 2 Contributors involving turbine & turbine pump, plus: 1) Ultimate loss of turbine because of inadequate lube oil cooling from AC operated lube oil cooling water circulating pump.	Case 1 and 2 Contributors involving turbine & turbine pump, plus: 1) Ultimate loss of turbine pump because of lack of bearing cooling water-supplied from AC cooling water pumps.	Case 1 Contributors plus: 1) AC dependence of all AFWS valves prohibits initial AFWS mission success.	Case 1 Contributors plus: 1) AC dependence of valves necessary for system actuation prohibits initial AFWS mission success.	Case 1 Contributors pertaining to turbine and turbine pump plus: 1) Potential failure of MSV6 because of loss of air leading to degraded steam supply and/or turbine overspeed trip.

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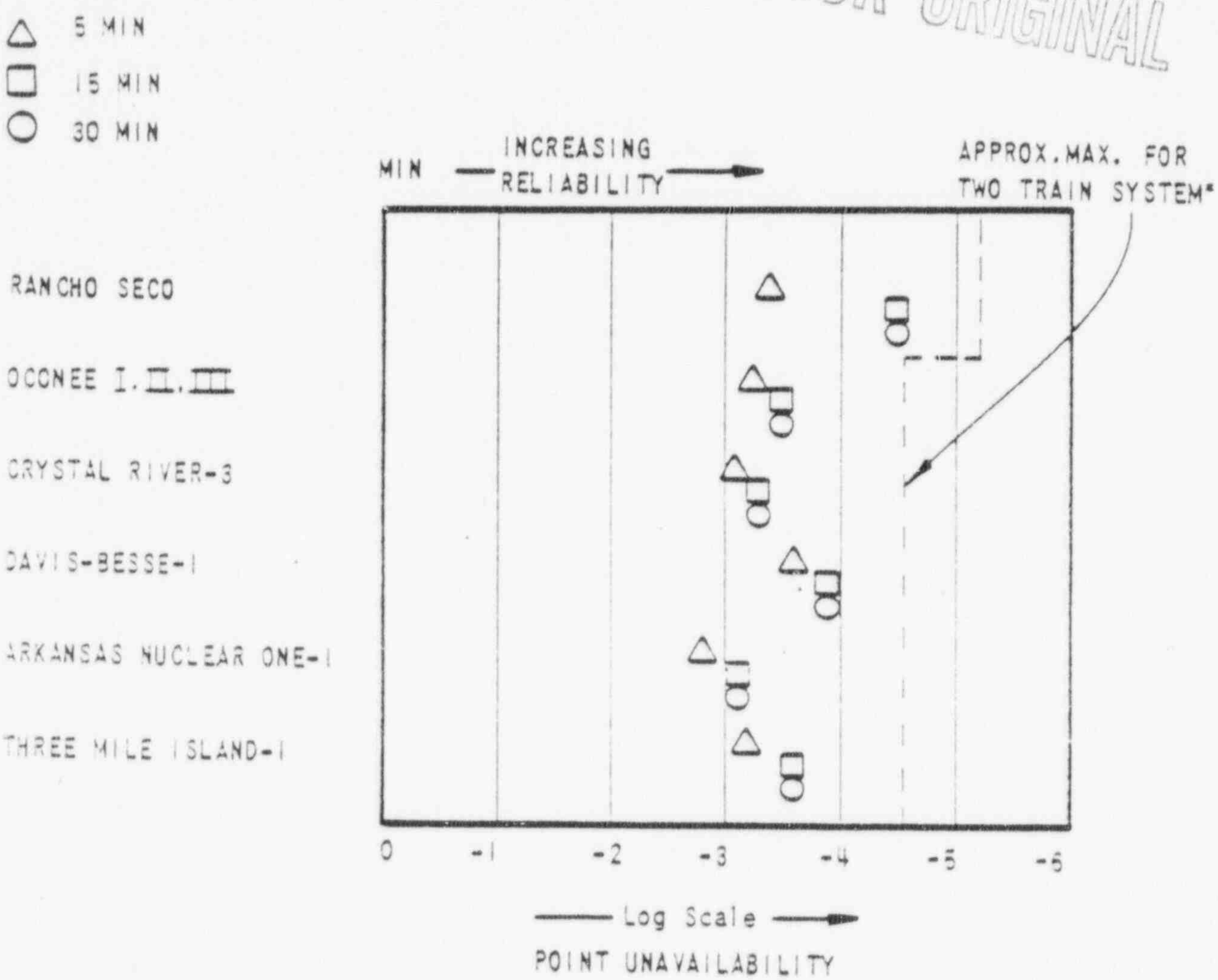

*UPPER LIMIT IS DIFFERENT FOR RANCHO SECO BECAUSE OF THE MULTI-DRIVE PUMP.

FIG. 1A RELATIVE AFWS RELIABILITIES, LMFW

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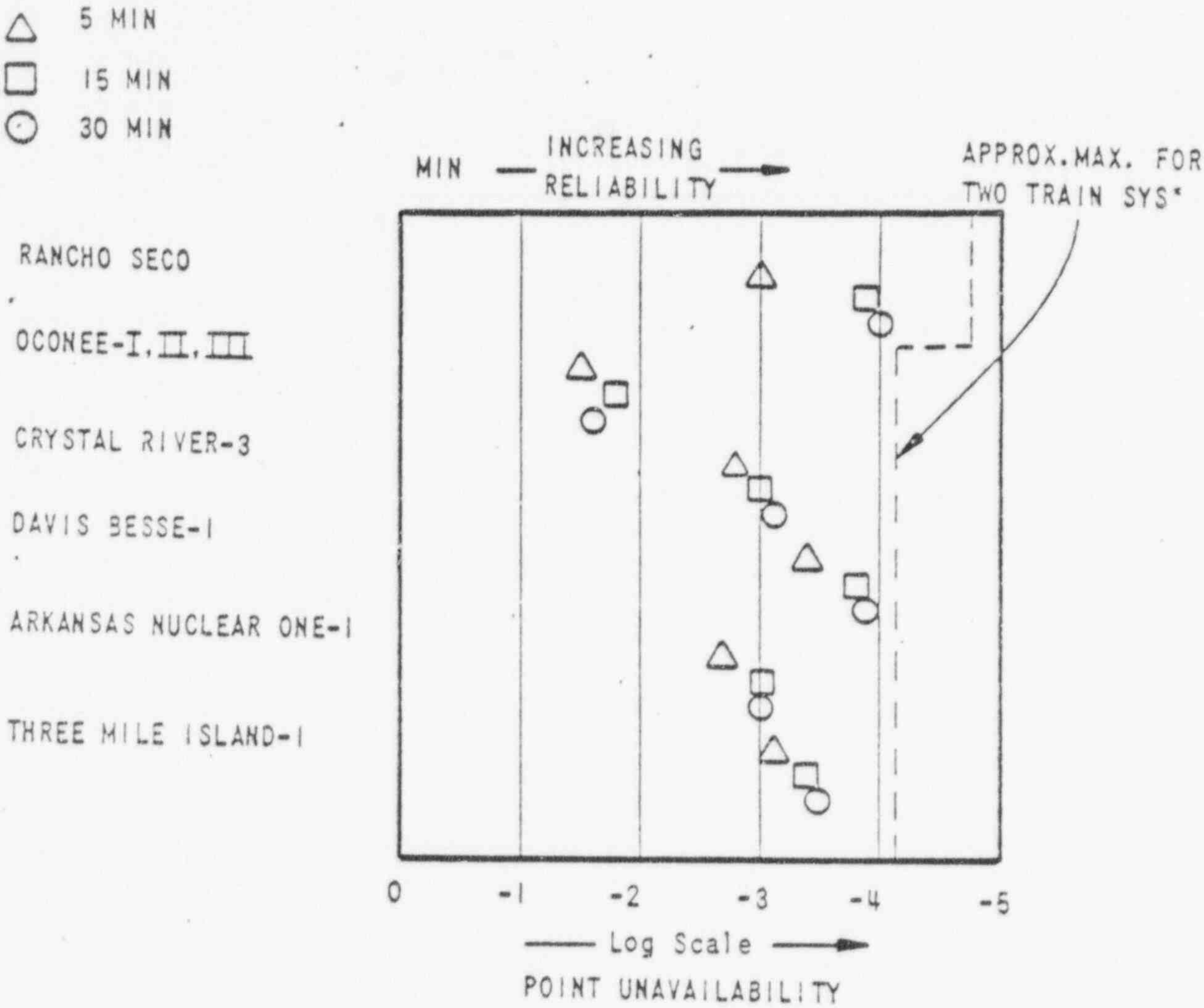

*WHERE ONE TRAIN IS ELECTRIC POWERED FROM A DIESEL GENERATOR (IE.,EXCLUDING DAVIS-BESSE-1). LIMIT IS DIFFERENT FOR RANCHO SECO BECAUSE OF THE MULTI-DRIVE PUMP.

FIG. 1B RELATIVE AFWS RELIABILITIES, LMFW/LOOP

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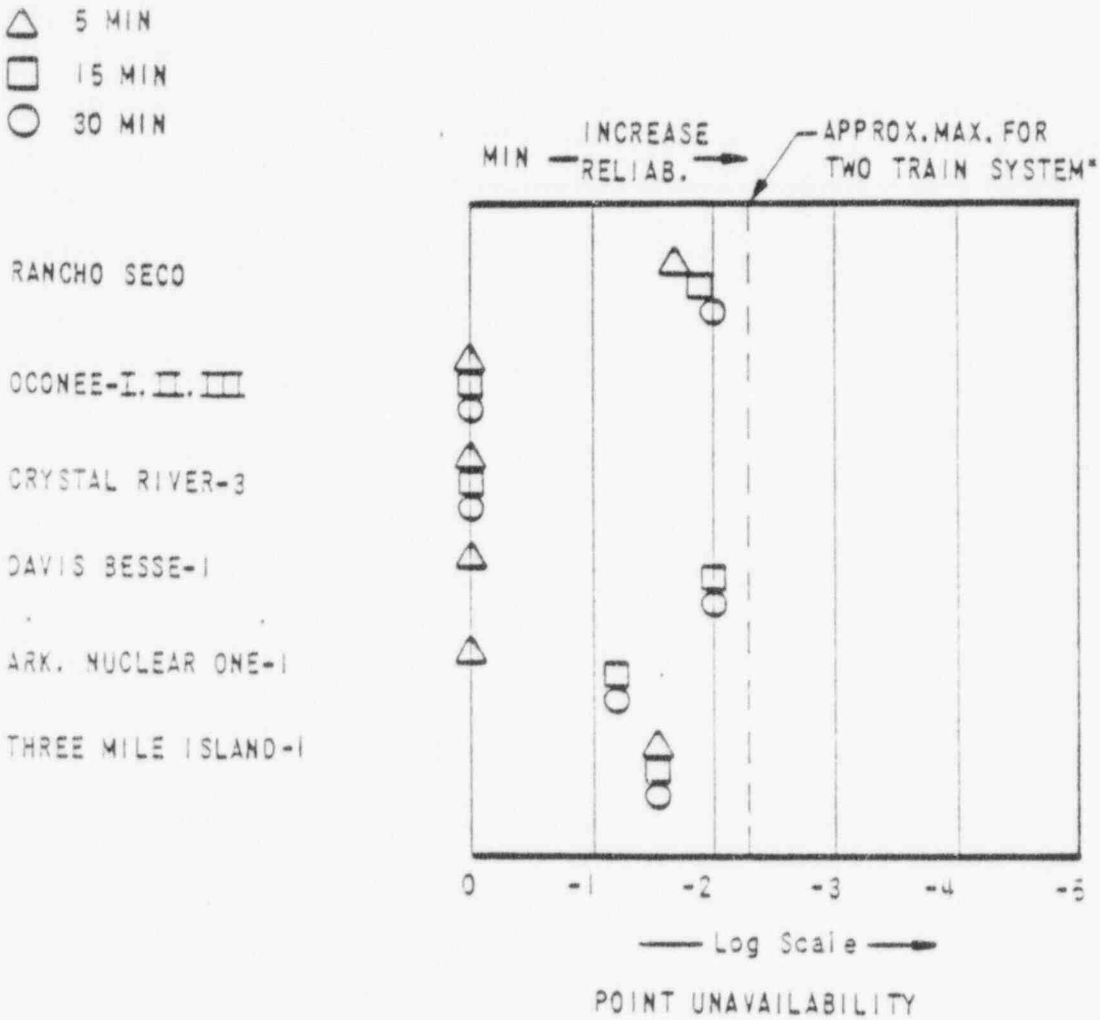

*WHERE ONE TRAIN IS ELECTRIC POWERED FROM A DIESEL GENERATOR (IE., EXCLUDING DAVIS BESSE-1)

FIG. 1C RELATIVE AFWS RELIABILITIES, LMFW/LOAC

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REFERENCES

1. DRAFT version of Appendix III (W), Auxiliary Feedwater Systems as transmitted in a letter from T. E. Murley (NRC) to E. A. Womack (B&W) November 8, 1979.

2. "Nuclear Power and Public Risk", IEEE SPECTRUM - Pgs. 58-74 - November, 1979.

3. WASH-1400 (NUREG-75/014), "Reactor Safety Study" USNRC, October 1975.

4. "Auxiliary Feedwater System Reliability Analysis for the Rancho Seco Nuclear Generating Station - Unit no. 1" Babcock & Wilcox, Sept. 10, 1979.

5. "Emergency Feedwater System Reliability Analysis for the Oconee Nuclear Generating Station, Unit No. I, II, III" Babcock & Wilcox, Revision 1, November 1979.

6. "Auxiliary Feedwater System Reliability Analysis for Crystal River Unit No. 3" Babcock & Wilcox, October 1979.

7. "Auxiliary Feedwater System Reliability Analysis for the Davis-Besse Nuclear Generating Station Unit No. 1" Babcock & Wilcox, Revision 1, November 1979.

8. "Emergency Feedwater System Reliability Analysis for Arkansas Nuclear One Generating Station Unit No.1" Babcock & Wilcox, Revision 1, November 1979.

9. "Emergency Feedwater System Reliability Analysis for the Three Mile Island Nuclear Generating Station Unit No. 1" Babcock & Wilcox, Revision 1, Dec. 1979.

10. "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" Volume 1, May 7, 1979, Babcock & Wilcox.

11. "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant", Volume III - Raised Loop Plants (Davis-Besse) May 16, 1979, Babcock & Wilcox.

APPENDIX A

NRC-SUPPLIED DATA USED FOR PURPOSES OF CONDUCTING A COMPARATIVE ASSESSMENT OF EXISTING AFWS DESIGNS & THEIR POTENTIAL RELIABILITIES

	Point Value Estimate of Probability of* Failure on Demand
I. Component (Hardware) Failure Data	
a. Valves:	
Manual Valves (Plugged)	$\sim 1 \times 10^{-4}$
Check Valves	$\sim 1 \times 10^{-4}$
Motor Operated Valves	
• Mechanical Components	$\sim 1 \times 10^{-3}$
• Plugging Contribution	$\sim 1 \times 10^{-4}$
• Control Circuit (Local to Valve)	
w/Quarterly Tests	$\sim 6 \times 10^{-3}$
w/Monthly Tests	$\sim 2 \times 10^{-3}$
b. Pumps: (1 Pump)	
Mecnanical Components	$\sim 1 \times 10^{-3}$
Control Circuit	
• w/Quarterly Tests	$\sim 7 \times 10^{-3}$
• w/Monthly Tests	$\sim 4 \times 10^{-3}$
c. Actuation Logic	$\sim 7 \times 10^{-3}$

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*Error factors of 3-10 (up and down) about such values are not unexpected for basic data uncertainties.

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II. Human Acts & Errors - Failure Data:

	← Estimated Human Error/Failure Probabilities → ← Modifying Factors & Situations →					
	With Valve Position Indication in Control Room		With Local Walk-Around & Double Check Procedures		w/o Either	
	Point Value Estimate	Est on Error Factor	Point Value Estimate	Est on Error Factor	Point Value Estimate	Est on Error Factor
A) Acts & Errors of a Pre-Accident Nature						
1. Valves mispositioned during test/maintenance.						
a) Specific single valve wrongly selected out of a population of valves during conduct of a test or maintenance act ("X" no. of valves in population at choice).	$\frac{1}{20} \times 10^{-2} \times \frac{1}{X}$	20	$\frac{1}{20} \times 10^{-2} \times \frac{1}{X}$	10	$10^{-2} \times \frac{1}{X}$	10
b) Inadvertently leaves correct valve in wrong position.	$\sim 5 \times 10^{-4}$	20	$\sim 5 \times 10^{-3}$	10	$\sim 10^{-2}$	10
2. More than one valve is affected (coupled errors).	$\sim 1 \times 10^{-4}$	20	$\sim 1 \times 10^{-3}$	10	$\sim 3 \times 10^{-3}$	10

II. Human Acts & Errors - Failure Data (Cont'd):

\- Estimated Human Error/Failure Probabilities -

	Time Actuation Needed	Estimated Failure Prob. for Primary Operator to Actuate AFWS Components
B) Acts & Errors of a Post-Accident Nature		
1. Manual actuation of AFWS from Control Room. Considering "non-dedicated" operator to actuate AFWS and possible backup actuation of AFWS.	$\sim$5 min. $\sim$15 min. $\sim$30 min.	$\sim 5 \times 10^{-2}$ $\sim 1 \times 10^{-2}$ $\sim 5 \times 10^{-3}$

III. Maintenance Outage Contribution

Maintenance outage for pumps and EMOVS:

$$Q_{Maintenance} \approx \frac{0.22 \text{ (\# hours/maintenance act)}}{720}$$

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APPENDIX B

COMPARABILITY WITH NRC ANALYSES FOR THE RELIABILITY OF AUXILIARY FEEDWATER SYSTEMS

B.1 Background

A major objective, established at the outset of B&W's Auxiliary Feedwater System Reliability Study, was the production of reliability results which could be compared with the results obtained by the NRC in its analyses of Westinghouse (W) and Combustion Engineering (CE) plants (References 1 and 2). The desired comparability was to be achieved by maintaining consistency with the NRC analyses; this consistency was to involve use of the same three event scenarios, the same fault tree analysis method, and the same assumptions, levels of detail and data employed by the NRC. Questions regarding the NRC's approach were to be resolved by direct consultation with NRC staff personnel who had participated in the W and CE analyses.

B&W did not have access to the fault trees used in the NRC study and therefore had to rely on telephone consultations with the NRC and independent engineering judgment in many cases. It is now evident to B&W that some inconsistencies have occurred which may invalidate a direct comparison between the B&W and NRC results. In particular, the NRC calculated reliabilities reported for some W plants are higher than would be possible using the B&W approach. This implies that systematic differences in the calculated reliabilities may reflect differences in the B&W and NRC approaches, and do not necessarily signify actual differences in system reliabilities.

B.2 Examples of Evaluation Approach Differences and Their Effects

One important area of difference between the NRC and the B&W approach involves an assumption concerning the number of operating pumps required to achieve mission success. It appears that, in some cases, the NRC gave credit for mission success upon successful operation of a single "half-capacity" pump. The effect of this on system reliability, depending on other areas of redundancy, is to shift reliability toward that of a three-train system.

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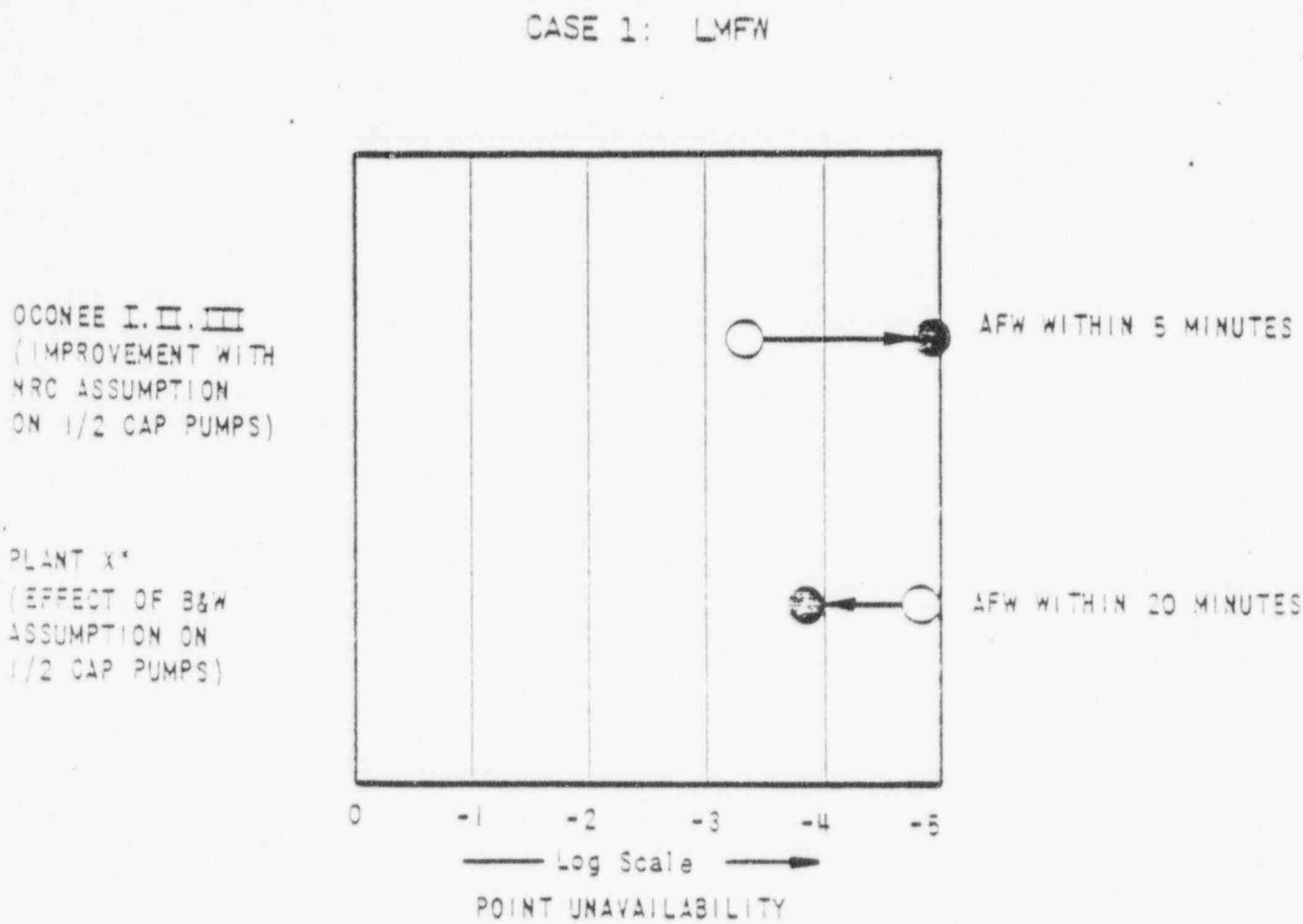
Two of the AFW systems analyzed by B&W also employed half-capacity pumps; however, B&W assumed that mission success could not be achieved by operation of one half-capacity pump by itself. An example of the effect of this assumption is shown in Figure B1 for the Oconee Units. As indicated in the figure, the assumption of mission success upon operation of a single half-capacity pump improves the calculated system reliability by more than an order of magnitude. An estimated reciprocal effect on one of the W plants analyzed by the NRC is also shown in Figure B1. As expected, the quoted reliability decreases by over an order of magnitude.

The use of different pump operation assumptions described above is a readily detectable difference between the B&W and NRC approaches; other differences may also exist. One such area of concern is the scope and level of detail of the fault tree analyses. The level of detail (fault tree failure rate data input level) used by B&W appears to be generally consistent with that used by the NRC; however, the scope (number of fault tree branches) of B&W's analyses may be greater. It is likely that, with more time available, B&W conducted a more comprehensive analysis; and a more comprehensive analysis frequently results in a lower calculated reliability.

B.3 Comparison of Reliability Results

Figure B2 shows a comparison of calculated reliabilities for the B&W operating plants with results obtained by the NRC for W and CE. The format for this figure was derived from References 1 and 2.

The figure demonstrates that, with allowances for analysis differences, the range of expected AFWS reliabilities for B&W plants is similar to that obtained by the NRC for W and CE.



*DATA OBTAINED FROM REFERENCE 1 AND PLANT X FSAR.

FIG. B1 EFFECT OF ASSUMPTION ON CALCULATED AFWS RELIABILITY

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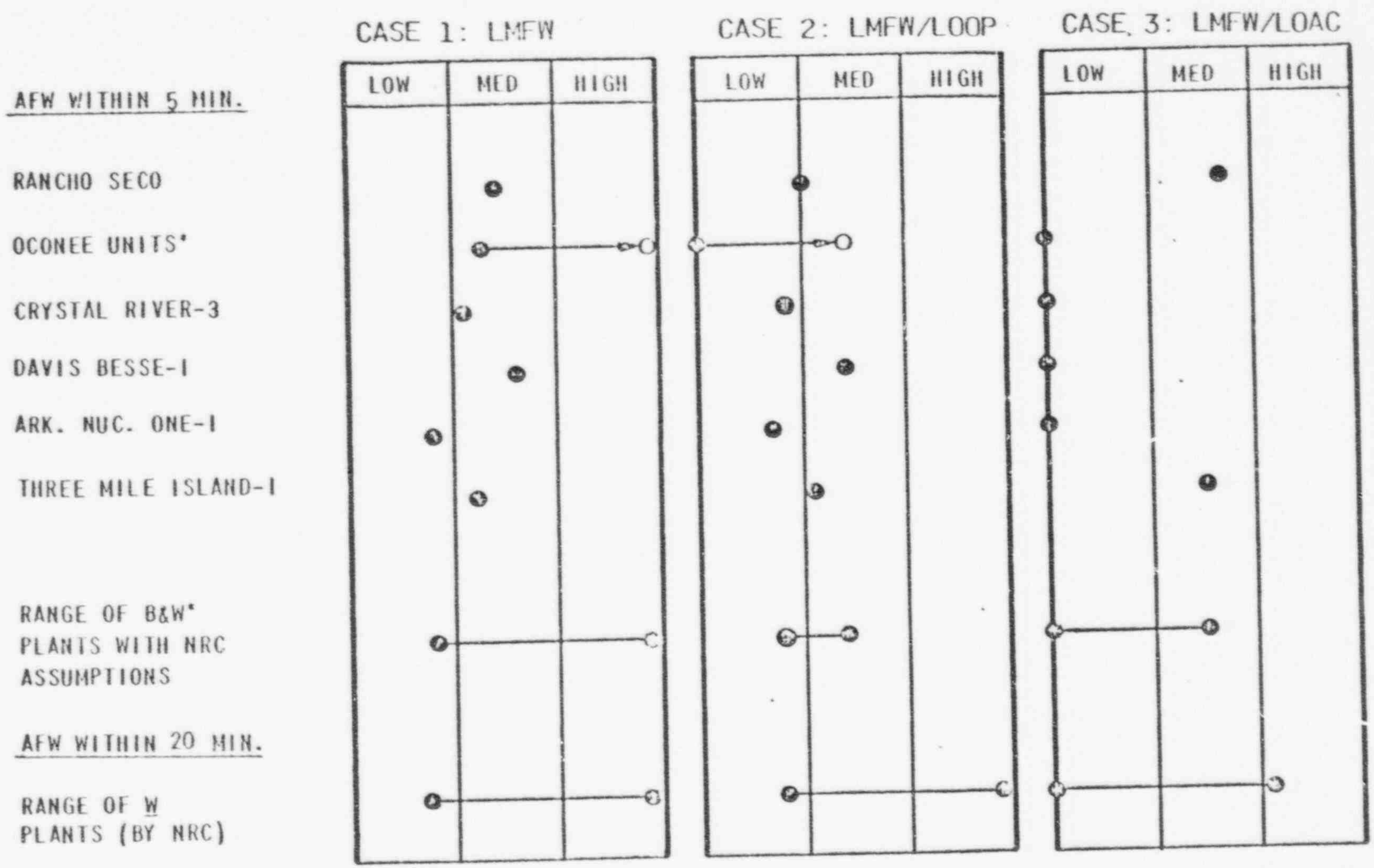

*RELIABILITY CHANGE DERIVED FROM FIG. B1

FIG. B2 COMPARISON OF B&W AFWS RELIABILITY WITH NRC RESULTS FOR W PLANTS

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