November 29, 2001

MEMORANDUM TO: John A. Zwolinski, Director, DLPM:NRR
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Thomas L. King, Director, DSARE:RES
Michael E. Mayfield, Director, DET:RES

FROM: Scott F. Newberry, Director Original signed by
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

SUBJECT: FINAL REPORTS: RELIABILITY STUDY: COMBUSTION ENGINEERING REACTOR PROTECTION SYSTEM, 1984-1998, NUREG/CR-5500, VOL. 10 AND RELIABILITY STUDY: BABCOCK & WILCOX REACTOR PROTECTION SYSTEM, 1984-1998, NUREG/CR-5500, VOL. 11

The final reports "Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998," NUREG/CR-5500, Vol. 10 and "Reliability Study: Babcock & Wilcox Reactor Protection System, 1984-1998," NUREG/CR-5500, Vol. 11, are attached for your information and use. These reports document analyses of the performance of the reactor protection system (RPS) in U.S. commercial pressurized water nuclear reactors (PWRs) designed by Combustion Engineering (CE) and Babcock & Wilcox (B&W).

These reports complete the development of a series of risk-based analyses using operating experience data for risk-significant systems, components and operating events. Attachment 1 lists the studies completed. These studies were part of our program of risk-based analysis of reactor operating experience to systematically identify risk-significant insights and provide feedback to the regulatory process.

The studies documented in these reports (1) estimate the B&W and CE RPS unavailabilities based on actual operating experience, including applicable test data, (2) compare these estimates with the estimates using data from probabilistic risk assessments and individual plant examinations, and (3) provide insights regarding failures and failure mechanisms associated with the operation of these systems and their associated trends. The reports include comprehensive treatments of the data uncertainties in the parameters estimated. They use Bayesian analyses for parameter estimation and incorporate data uncertainties explicitly in determining the significance of trends, comparing estimates with parameters from probabilistic risk assessments and individual plant examinations, identifying insights and making conclusions.

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These reports and similar system reliability studies conducted by the Office of Nuclear Regulatory Research (RES) support the strategic goals of maintaining safety; improving regulatory effectiveness, efficiency, and realism; reducing unnecessary burden; and increasing public confidence. The major findings that support each of these strategic goals follow, with specific cognizant organizations indicated in parentheses.

1. **Maintaining Safety** - These reports provide evaluations of the system unavailabilities and performance trends over time. These analyses of system performance trends in time should be useful for determining whether safety is improving, deteriorating, or remaining constant in light of both the agency and licensee safety initiatives. (SPSB:DSSA:NRR, IIPB:DIPM:NRR, REAHFB:DSARE:RES)

    a. *Overall system unavailability*.

        i. There are four basic designs of the RPS for CE plants. The mean unavailability for these designs varied from 6.5E-6 to 7.5E-6, with no credit given for manual trips by the operator. Credit for manual trips by the operator improves the RPS unavailability to a range of 1.6E-6 to 5.7E-6.

        ii. The calculated unavailability for CE plants shows little sensitivity to the type of the core protection calculator, i.e, analog or digital, used in the various designs.

        iii. There are two basic designs of the RPS for B&W plants (i.e., Oconee and Davis-Besse). The mean unavailabilities with no credit given for manual operator action are 7.8E-7 (Oconee design) and 1.6E-6 (Davis-Besse design). Giving credit for operator action improves the mean RPS unavailabilities to 8.7E-9 (Oconee design) and 8.4E-7 (Davis-Besse design).

    b. *Unplanned demand frequency trends*. Statistically significant decreasing trends in unplanned reactor trips were observed for both CE and B&W plants.

    c. *Failure frequency trends and common-cause failures (CCFs)*

        i. The trends in CE RPS component failure frequency and the number of CCF events decreased significantly over the study period.

        ii. The trends in component failure probabilities and number of CCF events were not statistically significant in the B&W data because the data are too sparse.

2. **Improving Regulatory Effectiveness, Efficiency and Realism -** The results, findings, conclusions, and information contained in these reports support a variety of risk-informed regulatory activities. These regulatory activities include plant inspections, technical reviews of proposed license amendments, regulatory effectiveness analyses, and development of risk-based performance indicators.

a. *Plant inspections*. The reports provide information for risk-informed inspection activities to enhance the use of inspection resources. The reports indicate the leading contributors to system unavailability and their failure causes which should be useful in the inspection program. In addition, they indicate the trends in demands and failure rates (see above) to assist in determining whether more, fewer, or the same level of RPS-related inspections are warranted. (IIPB:DIPM:NRR)

b. *Technical reviews of proposed license amendments*. The results of these studies can be used to compare licensees' RPS reliability estimates in risk-informed applications under Regulatory Guides 1.174, 1.175, and 1.177 with operating experience (see overall system unavailability above). These comparisons could be used to identify areas for further review where there may be substantial differences affecting the risk calculations in the submittal. (SPSB:DSSA:NRR)

c. *Regulatory effectiveness analyses*. The information in these reports can be used to determine whether the impact of the regulatory activities have achieved the intended risk result by comparing the goals with the observed experience. The trending information on demands and failures also provides information for determining the degree of change these activities may have accomplished. (REAHFB:DSARE:RES)

d. *Standardized Plant Analysis Risk (SPAR) Model Development*. The results of this work will be used to update the RPS performance data contained in the SPAR models, which are used by staff analysts in the performance of regulatory activities such as the Accident Sequence Precursor (ASP) program. (OERAB:DRAA:RES)

The technical insights that can be used to support this strategic goal include the following:

a. *Leading contributors to system unavailabilities*: Common-cause failures contribute greater than 99 percent to the overall unavailability of all CE and B&W designs. Automatic actuation of the shunt trip mechanism within the reactor trip breakers and maintenance improvements have resulted in performance improvements such that reactor trip breakers contribute less than one percent to the overall RPS unavailability of all CE and B&W designs.

b. *Failure causes*: Over all RPS designs studied for all components, the vast majority (80 percent) of RPS CCF events can be attributed to either normal wear or out-of-specification conditions.

c. *Failure trends*: The CE RPS component failure frequency and the number of CCF events decreased significantly over the study period. Trends in component failure probabilities and counts of CCF events were not statistically significant in the B&W data because the data are too sparse.

3. **Reducing Unnecessary Burden -** These reports include engineering insights that provide information that may be used in inspection activities to examine failure mechanisms consistent with their risk significance and, consequently, reduce unnecessary inspection burden. (Regional offices, IIPB:DIPM:NRR)

   The technical insights summarized under the “Improving Regulatory Effectiveness, Efficiency and Realism” strategic goal can also be used to reduce unnecessary burden by limiting activities in areas that are not important contributors to reliability or by adjusting intervals for inspection consistent with observed trends in performance. These include insights associated with leading contributors to system unreliabilities and failure causes noted above.

4. **Increasing Public Confidence -** The final analyses provide rigorous and peer-reviewed evaluations of operating experience to enhance the technical credibility of the agency with respect to quantitative risk assessment. Specifically, they demonstrate the agency’s ability to analyze operating experience independently of licensee-sponsored risk assessments. These independent assessments allow the agency to determine whether licensee assessments of risk are reasonable.

To help better identify and relate this detailed information to various risk-important regulatory applications, we have provided a Foreword section in these reports. The Foreword sections provide directions to the relevant quantitative and qualitative information contained in each report. The Foreword also indicates the appropriate type of engineering review of this information needed for application on a plant-specific basis.

In addition to the report insights noted above, a cooperative activity is underway between OERAB and the Inspection Program Branch (IIPB) of the Office of Nuclear Reactor Regulation (NRR) to make more effective use of pertinent insights and information from these and similar works associated with risk-informed inspection activities. OERAB is working with IIPB to develop and test a process to better capture risk-based operating experience and update risk-informed inspection activities using operating experience from system studies, component studies, common-cause failure database, and accident sequence precursor events. In this regard, OERAB is ready to assist users of these reports.

Drafts of these reports were provided earlier for peer review and comment to NRR, the regions, and industry and public interest organizations. These reports have been revised to incorporate the resolutions to the comments. Attachment 2 is a tabulation of the comments and the associated resolutions.

Attachments: As stated

cc: See next page

Multiple Addresses

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Risk-Based System/Component/Event Studies Produced by OERAB

NUREG/CR-5500:

Vol. 1: Auxiliary/Emergency Feedwater System, 1987-1995
Vol. 2: Westinghouse Reactor Protection System, 1984-1995
Vol. 3: General Electric Reactor Protection System, 1984-1995
Vol. 4: High-Pressure Coolant Injection (HPCI) System, 1987-1993
Vol. 5: Emergency Diesel Generator Power System, 1987-1993
Vol. 6: Isolation Condenser System, 1987-1993
Vol. 7: Reactor Core Isolation Cooling System, 1987-1993
Vol. 8: High-Pressure Core Spray System, 1987-1993
Vol. 9: High-Pressure Safety Injection System, 1987-1997
Vol. 10: Combustion Engineering Reactor Protection System, 1984-1998
Vol. 11: Babcock & Wilcox Reactor Protection System, 1984-1998

NUREG/CR-5496:

Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980-1996

NUREG/CR-5750:

Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995

NUREG-1715:

Vol. 1: Component Performance Study - Turbine-Driven Pumps, 1987-1998
Vol. 2: Component Performance Study - Motor-Driven Pumps, 1987-1998
Vol. 3: Component Performance Study - Air-Operated Valves, 1987-1998
Vol. 4: Component Performance Study - Motor-Operated Valves, 1987-1998

AEOD/S97-03:

Special Study Fire Events - Feedback of U.S. Operating Experience (1965-1985)

Resolutions of Comments Received on Draft Reliability Study Reports of Combustion Engineering Reactor Protection System, 1984-1998 and Babcock & Wilcox Reactor Protection System, 1984-1998

Copies of the draft reports titled “Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998" and “Reliability Study: Babcock & Wilcox Reactor Protection System, 1984-1998" were provided to both internal and external stakeholders for review. The following stakeholders indicated that they had no comments on either reactor protection system (RPS) reliability study: Combustion Engineering Owners Group, Westinghouse Owners Group, Electric Power Research Institute (EPRI), Union of Concerned Scientists (UCS), and the Division of Systems Analysis and Regulatory Effectiveness in the NRC Office of Nuclear Regulatory Research. The stakeholders that provided comments on the reports were: Babcock & Wilcox Owners Group (BWOG) [B&W report only], Science Applications International Corporation (SAIC), University of Maryland (UofMD), the Division of Engineering Technology in the NRC Office of Nuclear Regulatory Research (DET:RES), the Plant Systems Branch of the Division of Systems Safety and Analysis in the NRC Office of Nuclear Reactor Regulation (SPSB:DSSA:NRR), the Electrical and Instrumentation and Controls Branch of the Division of Engineering in the NRC Office of Nuclear Reactor Regulation (EEIB:DE:NRR), and the Divisions of Reactor Safety in NRC Region I (DRS:RGN-I) and Region III (DRS:RGN-III).

Two notable technical comments were received. The first pointed out that the number of reactor trip breakers and their configuration in the Oconee RPS design in the Babcock & Wilcox (B&W) study were incorrect and that the Oconee design also has silicon-controlled rectifier (SCR) trip capability of the regulating rod groups. Resolution of this comment has resulted in modifications to the fault trees and cutsets for the Oconee design in the B&W report. The modifications have been reflected in all of the associated unavailability calculations. Because the modifications involved changes in the number of reactor trip breakers in the Oconee design, the number of trip breaker demands changed. Since industry pooled data was utilized for the number of breaker demands, this change in the number of industry demands required that all affected calculations in the CE report also be requantified.

The second notable technical comment was that the use of a 50 percent rod failure criterion for both studies was too high. The rod failure criterion for both studies has been lowered to 20 percent. This change is consistent with the previous study of the Westinghouse RPS study. It is also consistent with the statement of considerations for 10 CFR 50.62 *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants*. Appendix G in both reports details the results of sensitivity studies of the rod failure criterion and the overall affect on the RPS unavailability calculations.

Resolutions of the remaining comments primarily resulted in discussion expansions and clarifications. The specific comments and the associated resolutions are detailed in the following tabulation. The tabulation includes a comment tracking identification number, the commenting organization, the specific comment, and the resolution details.

ID	Commenting Organization	Specific Comment	Resolution Details
1	BWOG	The description of the reactor trip breakers (RTBs) for the Oconee reactor protection system (RPS) design is incorrect. The Oconee design has six RTBs, not four as indicated on page 3. There are two AC breakers and four DC breakers, in the Oconee design. The pairs of DC breakers are not mechanically connected as indicated on page 3 and in the note attached to Table A-2. However, each DC breaker is of a two-pole design; the two poles on each breaker are connected, such that a trip of any one DC breaker will interrupt half of the holding power to two safety rod groups. NUREG-1000 provides a good description of the RTB configuration: "Power is supplied to the control rod drive mechanisms as follows (see Figure 3.4). Power to each group of safety rods is provided via two separate sources (a main supply and a secondary supply). The supply line from each source contains one ac reactor trip breaker in series with one dc reactor trip breaker. To interrupt power to a given group of safety rods requires that either the ac or the dc breaker in each of the two paths open. Each dc trip breaker in the B&W design is a two-pole device supplying power to two safety rod groups, such that the failure of one dc trip breaker in conjunction with the failure of its associated (upstream) ac trip breaker will result in power still being supplied to two out of the four safety rod groups, hence maintaining them in the withdrawn position." Using the stated mission success criteria that two of the four safety rod groups must trip to ensure shutdown, mission failure requires coincident failure of three RTBs. Consequently, the model and results as presented in the report are incorrect. The top cutset shown on table F-1, common cause failure of two trip breakers, which comprises 52% of the system failure probability, is in fact a success because two safety rod groups will trip (and also the three regulating groups).	a. Using the design information detailed in NUREG-1000, the fault trees and cutsets for the Oconee design in the B&W report have been modified and requantified to reflect the corrected configuration, including credit for the electronic silicon-controlled rectifier trip capability of the regulating rod groups. b. For consistency with the statement of considerations for the ATWS rule (10 CFR 50.62), the mission success criteria has been revised from two of the four safety rod groups to 20 percent of the safety rods. All calculations have been updated to reflect this change. b. All calculations in either report that use industry pooled data, such as reactor trip breaker demands, have been requantified as required to reflect the corrected Oconee design which now contains six reactor trip breakers rather than four as analyzed originally. c. All affected figures, tables and discussions in both reports have been updated.

ID	Commenting Organization	Specific Comment	Resolution Details
2	BWOG	The Oconee design also has electronic silicon controlled rectifier (SCR) trip. Even if there where a triple failure of a selected three RTBs (i.e., failure of AC breaker A and both DC breakers C, or failure of AC breaker B and both DC breakers D) the regulating rod groups would trip via the SCRs. There is conservatism in the analysis because the mission success definition, which credits only the safety rod groups, does not credit the regulating rod groups (i.e., groups 5, 6, and 7). These rod groups trip via the AC trip breakers and/or by removal of SCR gating power. If a realistic mission success definition were to be used (i.e., one that credits the three regulating rod groups as approximately equal to two safety groups with respect to negative reactivity), then mission failure would require coincident failure of four trip devices (i.e., the AC breaker, both DC breakers, and the electronic SCR trip) associated with the same power source. It is the use of conservative mission success criteria, combined with the different implementations of the electronic SCR trip, which accounts for the difference in results between the Oconee and Davis-Besse RPS designs. This is discussed in BAW-10167A (the mission success criteria used in BAW-10167A was also conservative), to explain the difference in results there. If a more realistic mission success definition was to be used (e.g., trip of two safety rod groups or three regulating rod groups), then both designs would yield similar results. The only important difference between the two implementations of the electronic SCR trip is that it will trip seven rod groups at Davis-Besse, and only three rod groups at Oconee, either of which is sufficient to shut down the reactor.	a. As noted in the resolution to Comment 1, the B&W study has been revised to include credit for the electronic silicon-controlled rectifier trip capability of the regulating rod groups. b. As a consequence of all of the modifications incorporated into the B&W study, the computed unavailabilities for the two designs when no credit is given for manual operator action are very similar.
3	BWOG	Appendix G (sensitivity study) fails to address the sensitivity to a very significant assumption, which is the assumption not to credit the regulating rod groups. For the reasons discussed above, there are statements in the report that are misleading. For example, that the benefit of manual trip in the Oconee design is limited by breaker common cause failure (page 38, third paragraph), and that there is “lack of redundancy in the diverse trip” (page 38, insight number 1). These statements are a product of modeling assumptions (mission success criteria) rather than system design.	a. As noted in the resolution to Comment 1, the B&W study has been revised to include credit for the electronic silicon-controlled rectifier trip capability of the regulating rod groups. b. The text in Appendix G has been modified to clarify any statements that could be misleading.

ID	Commenting Organization	Specific Comment	Resolution Details
4	BWOG	In addition, each B&W plant has implemented a diverse scram system (DSS) to comply with the ATWS Rule (10CFR50.62), that has not been credited in this study. The implementation of DSS is plant-specific, but the general design is that it senses wide-range reactor coolant system pressure (i.e., separate sensors from those used in the RPS) and trips the reactor through means diverse from the RTBs and electronic trip of the RPS. This is performed by interrupting power to the Control Rod Drive Control System, thus de-gating the SCRs by means diverse from the RPS de-gating contacts. The statement on page 2 of the report, that "these systems use diverse trip parameters but use the same trip breakers" is incorrect. None of the B&W plant DSSs use the RTBs, as that would be contrary to the requirements of the ATWS rule.	Section 1 of the B&W report has been corrected to indicate that the diverse scram system removes gating power to the silicon-controlled rectifiers through separate relays
5	BWOG	The active shunt trip enhancement was added to the RTBs as a post Generic Letter 83-28 improvement. The benefit of the shunt trip is that it applies much more force (torque) to the trip shaft than the undervoltage device, which relies upon stored energy. In surveillance testing, the undervoltage and shunt trip devices are tested separately. Breaker mechanical failures (e.g., armature binding, sluggish operation) that result from failure of the undervoltage device test are reported as breaker failure. In most cases however, the shunt trip would have tripped the breaker in actual service. However, it is sometimes difficult to establish from the failure descriptions that this was the case. The few events from the operating history that were assigned to breaker mechanical (BME) are probably more accurately attributed to the undervoltage device. At the very least, it should be noted that the failure rate for the breaker mechanical (BME) mode is conservative.	The breaker mechanical failures were reviewed and the previously identified failure for B&W was reclassified as an undervoltage failure since the shunt trip would have tripped the breaker. Since this failure data was pooled across both the B&W and CE reports, the computations in both reports were revised and all affected information has been updated.

ID	Commenting Organization	Specific Comment	Resolution Details
6	BWOG	The summary of operating experience in Table B-1 indicates two independent non-failsafe failures of trip logic relays in 1986 and 1991. We can identify no NPRDS or LER reports that involve failure to trip of any reactor trip module relay (either main trip relay or trip logic relay) during this time. The relays used in the reactor trip modules (i.e., main trip relay and trip logic relays) are of failsafe design. These relays are operated with the coils normally energized and the contacts closed. The coils de-energize and the contacts open to trip the reactor. Due to the failsafe design, the likelihood of failure to trip is very small relative to other failure modes. The majority of failures in B&W plant as well as general operating experience for normally-energized (contact open-to-trip) relays involve either spurious open or failure to reset (close). Our review of the NPRDS and LER database indicates only one report of a reactor trip module relay failing to trip, and that was a partial failure. In that case, one of the four contacts on a relay failed to open, which reduced the redundancy in a single RPS channel from 2-out-of-4 to 2-out-of-3. This scarcity of failures makes it difficult to estimate a failure rate. However, the scarcity of failures has also impacted the calculation of common cause failure rate, resulting in the conditional common cause failure probability for relays (Table E-10) being essentially based upon generic component data (Table E-5). The generic impact vector in Table E-5 assigns a conditional common cause failure probability (sometimes called a Beta-factor) of about 0.08. This is an extremely conservative and pessimistic treatment for the reactor trip module relays, given their failsafe design. The main trip and trip logic relays are high quality sealed devices; they are not prone to environmental stresses (such as contaminants) and require no human adjustment or calibration. It is inappropriate to assign these components conditional common cause parameters that are based upon the general component population. It is especially difficult to find an engineering basis for the assignment of large order common cause failures to these components.	a. The two independent non-failsafe failures of trip logic relays cited in the B&W RPS study were identified through our analysis of LER and NPRDS records. The 1986 event occurred on 1/14/86 and was documented in LER 28986003. This report described a relay to the shunt trip failing to operate. The 1991 event occurred on 9/15/91 and was documented in an NPRDS failure record. This report described a relay in the manual bypass module hanging up. Both of these failures adversely affected the ability of the relays to perform their safety functions. b. As shown in Table 3-2, there were no CCF failures identified for the relays. Thus, the CCF basic event probabilities for the relays were based on prior distributions. Since there is not sufficient data on relays by themselves to populate the prior, the prior is conservatively based on all RPS components. The prior values were reassessed and the resulting CCF basic event probabilities are nominally lower than originally estimated (dropped from .08 to .02).

ID	Commenting Organization	Specific Comment	Resolution Details
7	BWOG	The number 5 and 6 cutsets for Davis-Besse (Table F-9) are incorrect. These cutsets (common cause failure of trip breakers and RPS channel C or D in maintenance) do not result in mission failure. These are in fact mission success because all rod groups will trip due to the electronic SCR trip. Bypass (test/maintenance) of an RPS channel does not affect the ability of SCRs to trip. When an RPS channel is placed in bypass, its associated main trip relay is bypassed. However the trip logic relays (coincidence logic) in that channel are still functional (in a reduced 2-out-of-3 logic). There is no bypass for the trip logic relays.	The fault trees and associated cutsets in the B&W report for the Davis-Besse RPS design have been modified to recognize that bypass (test/maintenance) of an RPS channel does not affect the ability of the SCRs to trip and that when an RPS channel is placed in bypass, the trip logic relays in that channel are still functional in a reduced 2-out-of-3 logic. The computations were revised and all affected report information has been updated.
8	BWOG	Table B-3 lists failure events at B&W plants that were considered to be common cause failure. Our review of the referenced events indicates the following: The events of 8/12/92 and 8/14/92 are not common cause failure. These are repeat failures of the same device. Both involve failure to trip of the channel B high temperature bistable. For the first event the cause was unknown and the bistable was replaced; however on the second failure the cause was discovered to be a bad connector on a module upstream of the bistable. The events on 5/25/87 and 5/11/87 appear to be unrelated except for coincidentally happening around the same time. The high temperature bistable failure (out of spec.) was caused by a bad component on a circuit board, which was replaced. The high pressure bistable suffered from setpoint drift, which was corrected by recalibration. These events did not share a common cause.	The CCF events for the B&W report were reanalyzed. Of the eight events originally listed, the events associated with bistables on 05/11/87, 05/25/87, 08/12/92 and 08/14/92 were reclassified as independent failures and deleted from the CCF tabulation. Thus, the original eight events was revised to four. All computations have been updated to reflect this change.
9	BWOG	The remaining events identified as common cause failures in Table B-3 involve small degradations of the trip signal (rather than complete inoperability), and were limited to a single trip parameter. The evidence to support common cause grouping of bistables across trip parameters (e.g., high pressure and high temperature) appears weak. It is recommended that the common cause grouping be limited to the four bistables with each separate trip parameter. At the very least, the coupling of common cause across equipment of unlike plant parameters should be considerably weaker that within the same plant parameter.	We agree that the coupling of common cause across equipment of unlike plant parameters is weak. Therefore, a coupling factor of 0.1 has always been applied to these CCFs . This value indicates a recognition of the low significance to the strength of the impact vector for these events. Thus, no changes to the report have been made.

ID	Commenting Organization	Specific Comment	Resolution Details
10	DRS:RGN-III	It appears that there is an opportunity to share the information obtained from the reliability studies with the inspectors for use in conducting their inspection activities. Presently any insights gained from the reliability studies are not necessarily shared with the inspectors. These points could be very insightful to the inspectors. The addition of a simple one-page "pullout" attachment or "fact-sheet" with these insights could assist the inspectors during maintenance/surveillance activities both during the planning and inspection phases.	To help better identify and relate this detailed information to various risk-important regulatory applications, we have provided a Foreword section in these reports. The Foreword sections provide directions to the relevant quantitative and qualitative information contained in each report. The Foreword also indicates the appropriate type of engineering review of this information needed for application on a plant-specific basis. In addition, a cooperative activity is underway between the RES Operating Experience Risk Analysis Branch (OERAB) and the NRR Inspection Program Branch (IIPB) to make more effective use of pertinent insights and information from these and similar works associated with risk-informed inspection activities. OERAB is working with IIPB to develop and test a process to better capture risk-based operating experience and update risk-informed inspection activities using operating experience from system studies, component studies, CCF database, and accident sequence precursor events. In this regard, OERAB is ready to assist other users of these reports as well as or other operating experience reports as well.
11	DRS:RGN-III	One of the objectives of the component performance study was to compare results with estimates in probabilistic risk assessments (PRAs) and Individual Plant Examinations (IPEs). However, it appears that comparisons were made to initial plant IPEs and not current PRAs. While this comparison is of some benefit, it is not based on current licensee failure data. In most cases licensees are updating or intend to update their failure data following each operating cycle. In order to be useful and complete all references to the licensees IPE/PRA should be referencing the licensees latest and most current PRA update. It is realized that this information is not always available but should be utilized when available. In addition, better accessibility to this information should be developed to ensure the reliability studies are as up-to-date as possible.	The RPS studies compare the unavailabilities of systems calculated using operating experience with unavailabilities calculated using PRA/IPE data. The most recent updates of plant IPE/PRA analyses of RPS unavailability are not docketed and readily available. The information in these reports can be used for comparisons with updated PRA/IPE values as they are submitted to support requests for various regulatory activities.

ID	Commenting Organization	Specific Comment	Resolution Details
12	U of MD	Section 2.2 Paragraph 4 states that lower order CCF events are not modeled in the fault trees based on a review that such events would not have contributed significantly to the overall RPS unavailability. What was the nature of the review? Review of the event data? Numerical (upper bound) estimate of their frequencies? Conservative estimate of the total contribution of such events? The concern is that such events may actually be as important as some others in the current model, even if they do not make significant contribution to the total.	The decision was made to leave out these types of events during the development of the Westinghouse RPS reliability study, the first of the four RPS reliability studies. This decision was based on several factors, including modeling of lower order CCF events increases the complexity of the model and numerical calculations based on actual event data. Experience with the other RPS studies has validated this decision in that trial cutsets of the lower order CCF events have tended to have unavailabilities that were several orders of magnitude less than the first order CCF model.
13	U of MD	Appendix E, Section E-3.1.2, Mapping of Data: The methodology of NUREG/CR-5485 (Guidelines on Modeling Common- Cause Failures in Probabilistic Risk Assessment) recommends a two-step “mapping” of the CCF impact vectors if data from other plants (vendor) are used: 1) mapping to account for difference in CCCG size, and 2) mapping or adjustment for design differences. Once these steps are taken, a pseudo homogenous database is created for the plant (vendor) specific study, allowing pooling of the data. The RPS reports seem to have skipped the second step. Paragraph 5 of section A.2.1.1 may be cited as a possible justification. However the evidence from review of CCF events points to many engineering reasons for variation of the nature of causes and impacts of CCF events from plant to plant, contrary to the statement made in Section A.2.1.1.	The referenced procedure refers to performing a plant-specific analysis. These RPS studies are not plant-specific. Their main purpose is to be representative of a class of plants. Thus, Step 2 was not performed. Appendix E has been revised to clarify this per our conversation with the reviewer.
14	U of MD	a. Appendix E, page E-10, Development of Prior Distributions: It is not clear if both or just one of the two methods described (i.e., Section E-3.3.2.1 and E-3.3.2.2) are used to estimate the prior (generic) distributions of various alpha factors. Section E-3.4.1 mentions Dirichlet priors when referring to results in Table E-10. b. Appendix E, Figure E-1: It is not clear what is meant by the box: “Select Impact Vectors Appropriate to BE Equation” c. Appendix E, p. E.3 last line (also Figure E-2): Should the failure criterion of a train be stated as “failure of one of two channels” and not “specific failure of one of two channels”?	We agree that the original text was unclear on all of these points. These were not separate methods but steps in the same procedure. Appendix E in both reports has been extensively revised to clarify this and the other items per our conversations with the reviewer.

ID	Commenting Organization	Specific Comment	Resolution Details
15	U of MD	B&W Report, Table 5: Footnote (a): Assumption of “monthly channel testing ” is non-conservative (See similar footnote in Table 7 of the CE study where a conservative assumption is made in a similar case).	BAW-10167 “Justification for Increasing the Reactor Trip System On-Line Test Interval” was initially issued in May 1986. One of the modifications identified in this report was a change in the channel functional test schedule from monthly to a staggered 45 day interval such that each of the four channels would be tested every six months. A check of the RPS testing frequencies of all operating B&W reactors verified that none of the plants had adopted this change for the period covered by this report, i.e., 1984-1998. They all continued the monthly channel functional tests. Clarifications of this fact have been incorporated into the B&W report.
16	U of MD	Since CCF events dominate the results in both studies, some qualitative discussion about the nature of the observed RPS CCF events would be very appropriate.	Sections 3.3 and 4.3 of both reports have been revised to include qualitative discussions of the nature of the observed RPS CCFs. These revisions included more complete discussions of the causes of the events.
17	U of MD	One of the stated objectives of these studies is to develop a “data-driven” estimate of system failure probability in real demands. Some of the system studies in the past (e.g., HPI systems report) provided such industry-wide direct, Bayesian estimate. The RPS studies do not provide this type of estimate.	Section 3.1 initially discussed the industry-wide direct Bayesian estimate. This discussion was causing confusion with other commenting groups. Therefore, to avoid this confusion, the references to such industry-wide direct Bayesian estimates have been deleted from the report. The RPS unavailability values presented in the reports are based strictly on using the appropriate fault tree models.
18	U of MD	Another objective of the these studies as stated in the Introduction is to “compare the results with the assumptions, models, and data used in PRAs and IPEs.” This is difficult to achieve given the level of information available in the IPE reports. Particularly, since the actual fault trees are not provided in the IPEs, it is not clear to what extent comparison with the assumptions and models in the IPEs is possible.	Sections 3.3 and 4.3 of both reports have been revised to include comparisons with the assumptions and models used in other similar studies and the IPEs.
19	U of MD	In both reports, the terms “risk assessment” and “unavailability analysis” are used interchangeably. Examples are: (B&W Study) Section A-3, last paragraph, page C-4, paragraph 4, Table C-2, footnote c, and Table C-3 footnotes a and b. For these studies “unavailability analysis” is the correct technical term.	Both reports have been modified and the term “unavailability analysis” is used consistently.

ID	Commenting Organization	Specific Comment	Resolution Details
20	U of MD	CE Report, Page E-10, last paragraph: 49 CCF events is not really a small amount of data! This is contrast with a similar statement on page E-10 of the B&W study. Nevertheless, pooling data is justified when proper mapping is done (see Comment 13).	To assure the best stability for the prior, the CE CCF events were pooled and properly mapped with the B&W and Westinghouse data. Text in Appendix E of the CE report has been modified to reflect this.
21	U of MD	Section 4.3, line 5: Perhaps a better term for “Additional CCF” is “Partial CCF.”	For clarification, the text has been modified from “Additional” to “Other.”
22	U of MD	Section E-3.3, line 2: Replace “GE” to “CE” (or B&W).	Text has been corrected in both reports.
23	U of MD	For a reader who wants to know more about the CCF methodology used, adding the following reference to Section E-4 is helpful: “Guidelines on Modeling Common- Cause Failures in Probabilistic Risk Assessment” NUREG/CR-5485, Nov. 1998.	The recommended reference has been added to both reports.
24	UofMD	B&W Report, Figure 3, p.10: Designators for DC Trip Breakers seem inconsistent with those used in fault trees (pages D-5 and D-6).	Designators for the dc trip breakers have been corrected as necessary to assure consistency across the report.

ID	Commenting Organization	Specific Comment	Resolution Details
25	EEIB:DE: NRR	During the period 1984 through 1998, U.S. operating plants experienced no system-level RPS failures. During the period from 1984 through 1998, no total system failures were observed in 612 demands on the CE RPS, and in 231 demands on the B&W RPS. Therefore, unavailability results for the both RPS designs modeled at the system level provided a very little useful information. Additional unavailability information was obtained from component failure data. However, NPRDS contained no component independent failure data for 3 years (i.e., 1996-1998). This data could not be obtained from LERs because LERs do not include it unless a test failure involves the loss of a train or a channel. (Since 1984, the LER Rule, "10CFR 50.73" shifted the emphasis reporting away from single component failures to focus on significant events). Furthermore, control rod failures were reported in NPRDS only through March 15, 1994. We believe that insufficient data affects accuracy of results and using failure data from similar plants operating overseas would increase the accuracy of results.	a. As stated in the resolution to Comment 17, references to industry-wide direct Bayesian estimates have been deleted from the report. b. The lack of NPRDS data after 1995 was recognized. Comparative statistical tests were conducted on this data to that available for 1996-1998. These tests determined that there was insufficient data after 1995 to properly conduct the unavailability computations. Therefore, the unavailability values and associated component trends given in these reports covers only the period from 1984-1995. The titles of the reports have been changed to reflect this fact. However, the engineering analyses were conducted using the LER information available through 1998 in an attempt to determine if trends seen in earlier periods continued through 1998 or if new issues were emerging. c. Searches of the NPRDS failure records show no evidence of a change in control rod failure reporting after March 15, 1994. d. The limited number of failures during the large number of demands is a sufficient sampling for estimating the unavailability parameters. Additional data would primarily simply reduce the uncertainty levels.

ID	Commenting Organization	Specific Comment	Resolution Details
26	EEIB:DE: NRR	For each study, INEL evaluated the failure data to determine if the failures or inoperability impacted reactor safety. The report indicated that for some failures, INEL could not determine whether a failure impacted reactor safety or not, and these failures were classified as unknown. In addition, in some cases INEL could not determine if the failure was a complete failure, partial failure or not a failure. The data was divided in to nine categories, depending on whether the failure impacted safety or not and whether the failure was a total failure, a partial failure or not a failure. Data from only one category (categorized as not fail safe/complete failure -NFS/CF) was fully evaluated and only partial-data from the following 3 out of the remaining 8 categories was evaluated. (1) In the NFS/UC category, the failure was not fail safe, which mean failure had a safety impact but it was not known whether the failure was complete. We believe that if a failure could impact safety, it would be conservative to consider such a failure for the study although completeness of failure is not known. (2) In the UKN/CF category, the safety impact of the failure is unknown, but a complete failure occurred. We believe that if safety impact of a failure is not known, it is possible that the failure might impact the safety in some circumstances. Therefore, it would be conservative to include such a failure in the study. (3) In the category UKN/UC, the safety impact of the failure is unknown, and it was not known whether the failure was complete. If the safety impact of a failure is unknown, it might impact safety in some circumstances; therefore, it would be conservative to account such a failure in the study. Both reports indicated that although all data for failures categorized as NFS/CF, NFS/UC, UKN/CF and UKN/UC was used for the engineering analysis and the CCF analysis, only fraction of the above independent component failure events were used in the RPS unavailability studies. It is not clear why only a fraction of above events were considered and what criterion was used for selecting or rejecting failure data from these above three categories. Considering a fact that the RPS is supposed to perform its function on demand immediately (without any delay), and every component in the RPS has a definite design-role to play to support the RPS-mission, we believe that it should not be difficult to determine whether failure or degradation of any component will impact the reactor safety. Examples of components whose safety impact on failure could not be established in the studies were trip breaker mechanical components, trip breakers shunt trip coil,	As discussed and clarified with NRR staff, the treatment of the failure data provides realistic estimates of the number of the unknown events that are included in the analysis. To accomplish this, two weighting factors are computed and applied to three groups in question. The two weighting factors estimate (1) the probability that a failure is complete and (2) the probability that the safety function is lost. The completeness probability factor is 1.0 for the UKN/CF category since these data are known to be complete. For the NFS/UC and UKN/UC data the completeness factor is assumed to be 0.5. The probability that the safety function is lost is based on a simple Bayesian update of the known data. For the UKN/CF data, this is computed as [(NFS/CF)+0.5} divided by [(NFS/CF) + (FS/CF) + 1]. Similarly, for the NFS/UC and UKN/UC data, the factor is [(NFS/UC + .05} divided by [(NFS/UC) + (FS/UC) + 1]. An example of this method is detailed in Footnote b to Table C-2.

ID	Commenting Organization	Specific Comment	Resolution Details
27	EEIB:DE: NRR	The rod failure criterion selected to define rod-failure event in each study appears to be less conservative compared to previous studies. Past studies defined the rod failure event as the failure of three or more rods to insert. In the current studies, the criterion used was the failure of 20 or more rods out of 69 rods for the B&W RPS or failure of 50% or more out of 89 rods for the CE RPS rod failure events. Thus, in current studies, if 19 rods in the B&W RPS failed to insert on demand or 43 rods in the CE RPS failed to insert on demand; the event would not be considered a failure.	As noted in the resolution to Comment 1, the rod success criteria in both studies have been revised to be consistent with the statement of considerations for the ATWS rule (10 CFR 50.62), i.e., "Insertion of only about 20 percent of the control rods is needed to achieve hot, zero power provided that the inserted rods are suitably uniformly distributed." This change means that the rod failure criteria is now about 7 or 8 rods and makes these RPS studies consistent with the previous PWR RPS study (i.e., Westinghouse). Additionally, Appendix G in both studies includes the results of sensitivity analyses to other rod failure percentages.
28	EEIB:DE: NRR	The common-cause modeling in the RPS fault tree is limited to the events that fail enough components to fail that portion of the RPS. For both the CE RPS and the B&W RPS designs, the RPS trip logic is based on an any-two-out-of-four combination, which yields six trip combinations for four monitoring channel signals. It is not clear, why the model for CCF evaluation in both studies was based on a one-out-of-two-twice logic combination.	Appropriate sections of both reports have been revised to clarify that the channel logic is based on any 2-out-of-4 except for the reactor trip breaker portion which is based on 1-out-of-2/twice.
29	EEIB:DE: NRR	It appears that all the unplanned reactor trips involving channel components, where the process parameter (which originated the reactor-trip) could not be identified, were omitted from these studies and most of the estimates in this report are based on test data. Although, increasing the number of demands in a failures to demands ratio could yield non-conservative results, the staff believes that the absence of data for some unplanned reactor trips, combined with unquantified uncertainties in test failures data, make the results of this study inconclusive and biased. It is not clear how demands which did not approximate the conditions of studies were addressed.	As shown in Table C-3 of both reports, unplanned demands as well as test demands were used for several components in each study. These include manual scram switches, mechanical breakers and control rod drives & rods. For these components, the number of failures and the corresponding number of demands could be accurately estimated from the demand and test data. For other components, particularly the channel components, only test data was used since the number of operational demands associated with the operational failures could not be determined accurately. The information reported in LERs or NPRDS was not sufficiently detailed to identify the number of similar components that activated during the event.
30	EEIB:DE: NRR	Past studies, including EPRI-ATWS (1976), NUREG-0460 (1978), and WASH-1270 (1983) resulted in RPS unavailability for both B&W and CE RPS on the order of 7.0E-7 to 1.1E-4. WASH-1400 estimated unavailability to be on the order of 1.3E-5 (median), with three or more rod failures as the dominant contributor for the B&W and the CE RPS. The current report assumes 20 or more rod failures; therefore, the RPS unavailability value is less than that of the WASH-1400 report.	As noted in the resolution for Comment 27, the rod failure criteria for both studies has been revised to be consistent with the statement of considerations for the ATWS rule (10 CFR 50.62). Section 3.3 in both reports has been expanded to note the differences in the unavailabilities found in various studies, including WASH-1400 and PRAs and IPEs.

ID	Commenting Organization	Specific Comment	Resolution Details
31	EEIB:DE: NRR	One of the objectives of these studies was to compare PRA assumptions used in the previous studies to actual operating experience. After reading this report, it was not clear as to what was the outcome of such a comparison. It would have been helpful if INEL identified which assumptions of previous studies are correct and which needed modification.	Section 3.3 of both reports has been revised to provide the comparison of these operational data based studies with the assumptions and models used in other similar studies, including IPEs.
32	EEIB:DE: NRR	In addition, we believe that a comparison between results of these studies (which are generic in nature) with the IPEs results (which were specific to each plant) will not meet the objectives of these studies, and that the generic raw values of RPS unavailability derived out of the B&W and CE studies will be of little use in the staff's regulatory activities.	The RPS studies compare the unavailabilities of systems calculated using operating experience with unavailabilities calculated using PRA/IPE data. The most recent updates of plant IPE/PRA analyses of RPS unavailability are not docketed and readily available. The information in these reports can be used for comparisons with updated PRA/IPE values as they are submitted to support requests for various regulatory activities.
33	DRS:RGN-I	CE Page 2; "It should be noted that the RPS boundary for this study does not include ATWS mitigation systems added or modified in the late 1980s. For Combustion Engineering nuclear reactors, these systems use diverse trip parameters but use the same trip breakers." This statement does not appear to be true for Calvert Cliffs or Millstone Unit 2 (both CE plants). At both plants, the diverse ATWS system trips the MG set breaker contractors not the RPS breakers.	Section 1 of the CE report has been modified to reflect that the motor-generator set breakers are part of the ATWS system.
34	DRS:RGN-I	CE Page 17; "The control rod failure criterion was chosen to be 50% (or more) of the safety control rods fail to insert." Success of control rod insertion in the Calvert Cliffs IPE is 74 of 77 CEAs (basis provided in IPE). The Calvert Cliffs PRA analysis establishes this criterion based on RCS overpressure. A technical basis for selecting 50% as the success criteria should be defined in the report.	As noted in the resolution to Comment 27, both studies have been revised to be consistent with the statements of considerations for the ATWS rule (10 CFR 50.62) and now assume a 20 percent rod failure criterion, approximately 7 or 8 rods. In developing the ATWS rule, consideration was given to requiring extra safety valves to reduce the peak pressure in the reactor vessel. However, with the inclusion of the other changes required to meet the ATWS rule, the value/impact of this additional change was deemed unfavorable and, therefore, no design changes were required to meet the ATWS rule.
35	DRS:RGN-I	CE Page 35, Table 17, Millstone Unit 2, "The Millstone Unit 2 IPE was not available." The Millstone Unit 2 IPE is currently available and the unavailability for reactor trip given in IPE table A-2 is 1E-5, which is similar to the other CE plant assumptions. This information could also be included in Figure 11.	The applicable areas of the CE report have been revised to include this information.

ID	Commenting Organization	Specific Comment	Resolution Details
36	DRS:RGN-I	a. CE Page 44, “Four CCF and potential CCF events were identified for the period 1984 through 1998.” Table B-2 appears to indicate that there were 49 not 4 common cause failures. b. CE Page 44, “A decreasing trend was observed for the 64 events.” again it appears that there are 49 events. It’s not clear what the data source is for the 64 events.	a. The applicable areas of the CE report have been revised to reflect that there were actually 65 CCF events. b. The sources of the CCF data are described in Section 2.3.1.
37	DRS:RGN-I	CE Page D-93, event description “CLUTCH POWER SUPPLY BUSES FAIL TO DE-ENERGIZE” This event description is only valid for RPS group 1. For groups 2- 4 the event should be “CEDM COIL POWER SUPPLY BUSES FAIL TO DE-ENERGIZE.”	The applicable areas of the CE report have been revised to reflect this information.
38	SAIC	Page A-3 (fifth bullet) in both the B&W and CE reports states that all unplanned trips that are reportable are critical trips. We understand that all unanticipated ESF demands (including reactor trip demands) are reportable, whether the plant was critical or shut down. Suggest INEEL confirm trip reportability and revise statement(s) if necessary.	Trip reportability was confirmed and appropriate text revisions have been incorporated into each report.
39	SAIC	Footnote b. for Table C-2 in both reports is confusing and would benefit from an example that applied the approach.	Footnote b for Table C-2 in both reports has been completely revised to better explain how the total failure weighted averages were determined.
40	SAIC	For consistency with the CE analysis, basic events for the failure of individual manual scram switches should be added to the B&W fault trees in Appendix D (this is expected to have little effect on the analysis results).	Individual manual scram switches were added to the B&W model and included in the cutset analyses. Thus, both studies now include the manual scram switches and treat them the same.
41	SAIC	The first paragraph on page E-8 in both reports describes the calculation of the NFS ratio. The NFS ratios listed in Table E-2 appear to be inconsistent with this definition. INEEL should check that the NFS ratios were calculated and applied correctly in the analysis.	Appendix E has been revised extensively. As a part of these revisions, the explanation of this particular calculation method has been expanded and the associated table has been updated to correctly reflect this method.
42	SAIC	Section 3.4 of the B&W report discusses the use of two trip signals in the base case analysis and the impact on the results if three trip signals were assumed to be available. If only one trip signal is applicable for some initiating events, then a sensitivity analysis should be performed for the one-trip-signal case. This comment is also applicable to Appendix G in both studies.	There are no transients anticipated where the RPS would receive only one trip signal. This concept is consistent with that used in the previous studies of the Westinghouse and General Electric RPSs.

ID	Commenting Organization	Specific Comment	Resolution Details
43	SAIC	Tables 1 and 17 of the CE report list a Group 1 RPS failure probability of 8.6E-6, while Table 12 lists a failure probability of 7.6E-6 (the RPS segments in Table 12 sum to this value also). This inconsistency needs to be resolved.	Corrections as necessary have been incorporated into the CE report to provide report consistency.
44	SAIC	We have made this comment before, but it seems the reports still focus on the quantitative results and trends but there doesn't seem anything on whether the specific types of failures/causes and when the failures are being detected, are changing. It is recommended more of these qualitative insights be added; such insights may be more useful to readers/users of this information than just the quantitative trends.	Section 4.3 has been expanded to include more qualitative insights.
45	SPSB:DSSA: NRR	(B&W) Introduction Page 1. Last paragraph. Component or system failures causing spurious reactor trips or not affecting the shutdown function of the RPS are not considered in the report. Are the demands counted?	The Introduction has been revised to specifically note that spurious trips are included as demands.
46	SPSB:DSSA: NRR	(B&W) Page 2, Objective 1. States that the results will be compared to individual plant IPEs. The comparison is limited to the generic overview of IPE results. The CE report has a broader discussion.	Sections 3.3 and 4.3 of both reports have been revised to include comparisons with the assumptions and models used in other similar studies and individual plant examination reports. Qualitative discussions of the nature of the observed RPS CCFs are also included. These revisions included more complete discussions of the causes of the events.
47	SPSB:DSSA: NRR	(B&W) Table 2 – add columns (lines) to separate data?	The table format has been revised to improve its legibility.
48	SPSB:DSSA: NRR	(B&W) 2.1.3 System Operation. It appears that all sensor data is pooled. Is this sound? These transmitters/switches may vary greatly in operating principles and performance and by process.	Sensor data was segregated into different general categories such as temperature or pressure. Because of the sparsity of data, however, no attempt was made to break the temperature sensors or the pressure sensors into different operational types.

ID	Commenting Organization	Specific Comment	Resolution Details
49	SPSB:DSSA: NRR	(B&W) Figure 2, Page 8. The ATWS system and the Anticipatory Rector Trip System (ARTS) are not included in the study. Are they to be included as separate systems later?	The exclusion of the ATWS and the ARTS systems is consistent with the previous studies of the Westinghouse and General Electric RPSs. We have no current plans to produce similar reliability analyses of either the ATWS or ARTS.
50	SPSB:DSSA: NRR	a. (B&W) Table 5. Bistable, logic relays, and SCRs are shown as not continuously operating. Are not these components continuously energized? b. (CE) Table 8. Bistable, logic relays, trip relays (trip breakers?) are stated as not continuously operating. Bistable and relays are continuously energized. Define continuously operating.	The word "continuously" has been dropped from the table heading. The associated footnote has been revised to clarify that operating components are those whose safety function failures can be detected over time.
51	SPSB:DSSA: NRR	(B&W) Figure 5. Shows trip modules and trip relays – Table 5 references logic relays – "logic relay" is not referenced in the RPS diagrams. Is this a generic term for pooled data?	a. The relay nomenclature used in the simplified RPS diagrams was in error. This has been corrected such that it now corresponds with that in the table. b. "Logic relay" is a generic term used for pooled data.
52	SPSB:DSSA: NRR	(B&W) Table 5 Assumes monthly testing – is this correct for all data? Most plants have a channel functional test schedule of 45 days with all channels tested every 6 months (staggered test basis)?	As noted in the resolution to Comment 15, discussions with current resident inspectors confirmed that all of the B&W plants continue to perform monthly surveillances, including channel functional testing. The channel testing is on a staggered basis.
53	SPSB:DSSA: NRR	(B&W) Note that in the "monthly" test, the sensors are not tested - only the rack equipment. It appears that the study assumes that the sensor gets tested as well. The sensor gets a shift "channel check," which is essentially a cross comparison of sensor output among channels of a particular functional unit. This is not a comprehensive surveillance and detects gross failures only. See CE report for channel check discussion.	We agree that the monthly check is not a comprehensive surveillance, but it is capable of detecting gross sensor failures. More comprehensive tests are conducted during shutdown. The response to comment 71 contains a more detailed discussion of when shutdown tests were or were not included as well as the sensitivity analyses results that indicate that exclusion of these tests does not significantly affect the results. The treatment of sensor test data for the B&W study was consistent with that used in the other RPS studies. The appropriate text in the B&W report has been revised to be consistent with that in the other RPS reports.

ID	Commenting Organization	Specific Comment	Resolution Details
54	SPSB:DSSA: NRR	(B&W) Section 2.3.1. No Failure (NF) is classified, for example, as a trip setting slightly out of specification. By TS, if this was in the LER, then this was a failure or indeterminate. Will functionally operable meet the analysis assumptions?	The majority of the NF data is obtained from NPRDS reports. Degradations, such as TS violations, reported in an LER are classified as NF if the component is still functionally operable and system success is still met.
55	SPSB:DSSA: NRR	a. (B&W) Figure 5. Define "Uncertain Failure Components" b. (CE) Figure 9, Page 10. Uncertain failure components – define.	The term "Uncertain Failure Components" has been replaced by explanatory words to clarify this step in the analysis process.
56	SPSB:DSSA: NRR	(B&W) Section 3.1 States that there were no total system failures in 231 demands. Is this RPS failures or no functional unit failures?	Section 3.1 initially discussed the industry-wide direct Bayesian estimate. This discussion was causing confusion with other commenting groups. Therefore, to avoid this confusion, the references to such industry-wide direct Bayesian estimates have been deleted from the report. The RPS unavailability values presented in the reports are based strictly on using the appropriate fault tree models.
57	SPSB:DSSA: NRR	(B&W) Lack of data. LERs do not report individual component failure unless loss of function occurs, and the NPRDS database ended in 1996, and was voluntary. Available data is limited.	As noted in the resolution to Comment 25, the unavailability values and associated component trends given in these reports covers are based on primarily on NPRDS data over the period from 1984-1995. However, the engineering analyses were conducted using the LER information available through 1998 in an attempt to determine if trends seen in earlier periods continued through 1998 or if new issues were emerging.
58	SPSB:DSSA: NRR	(B&W) Section 4.3.1 References Figure 11 and states that no trend was seen among the 4 events. Figure 11 appears to depict more than 4 events.	The plotting routine used to generate this figure was changed to eliminate the display confusion. The new figure clearly shows the two CCF events remaining after the analysis discussed in the resolution to Comment 8.
59	SPSB:DSSA: NRR	(B&W) Section 4.3.2 References Figure 13 as "logic relays." Are these the trip relays shown in the RPS diagrams?	The referenced logic relays are the same as the trip relays in the RPS diagrams. The component code used in the diagrams was in error and has been corrected to help eliminate this confusion.
60	SPSB:DSSA: NRR	(B&W) Page 38, Fourth Paragraph. Report states that "…there were no total system failures in 231 demands…" "Total Trip System" failures?	Section 3.1 initially discussed the industry-wide direct Bayesian estimate. This discussion was causing confusion with other commenting groups. Therefore, to avoid this confusion, the references to such industry-wide direct Bayesian estimates have been deleted from the report. The RPS unavailability values presented in the reports are based strictly on using the appropriate fault tree models.
61	SPSB:DSSA: NRR	(B&W) Page A-3, First Bullet. Failures can also be that instrument drifts such that actuation will not occur, not just spurious actuations.	As noted in A.1.1, failures include losses at a component level that would contribute to loss of the safety function of the RPS. Reported instrument drifts that would have prevented actuation are included in the failure data.

ID	Commenting Organization	Specific Comment	Resolution Details
62	SPSB:DSSA: NRR	(B&W) Page A-8. Second Paragraph States that data differences were noticed between tests that were performed while operating and those performed while in shutdown. Was this difference noted for transmitters? Shutdown tests consist of calibration and functional tests – instruments could appear operable by channel check, but fail calibration or functional tests..	Table C.3 shows that for all of the components considered, failure probability differences could be identified only for the pressure sensor/transmitters. Following the convention of the earlier RPS studies, the subset of operations data was used in the calculations. Sensitivity runs were conducted that showed that this selection had a negligible impact on the overall RPS unavailability.
63	SPSB:DSSA: NRR	(B&W) Page B-1. Why is data prior to 1984 not used?	As discussed in report section A-1.1, events prior to 1984 were excluded for two reasons: (1) changes incorporated into the RPS as a result of the 1983 Salem ATWS event and (2) changes in the failure and event reporting requirements adopted in January 1984.
64	SPSB:DSSA: NRR	(B&W) Page B-1. In the list of components, "TLR" is not listed but shown on Tables B-1/B-1a. MSW is listed as a component but is not shown on the table.	The list of components has been updated. Some components listed, however, will not appear in subsequent tables because events with failure completeness (degradation) values less than 0.5 are excluded from the counts of independent events in Table B-1.
65	SPSB:DSSA: NRR	(CE) Page ix. The values given for RPS unavailability are significantly different from Table 1. Why the difference?	Corrections have been made to reflect updated results and to assure report consistency.

ID	Commenting Organization	Specific Comment	Resolution Details
66	SPSB:DSSA: NRR	(CE) Page xiii. Second Paragraph. Suggests that the report is not directly applicable unless additional review is performed on a plant specific basis. This limits the value of the report. How would plant specific data be integrated with the results of the report?	The information provided in these reports is directly applicable but additional plant-specific information after the study period would be useful. The reports indicate the leading contributors to system unavailability and their failure causes. This information could be applied directly to plant inspections in that plant-specific data could be compared to this generic information to see if new or different performance is occurring at the specific plant. In addition, the results indicate the trends in demands and failure rates. This could be used to assist in determining whether more, fewer, or the same level of RPS-related inspections are warranted. Other uses include comparing the results to licensees' RPS reliability estimates in risk-informed applications under Regulatory Guides 1.174, 1.175, and 1.177 with operating experience. These comparisons could be used to identify areas for further review where there may be substantial differences affecting the risk calculations in the submittal. The information in these reports can be used to determine whether the impact of the regulatory activities have achieved the intended risk result by comparing the goals with the observed experience. The trending information on demands and failures also provides information for determining the degree of change these activities may have accomplished.
67	SPSB:DSSA: NRR	(CE) Page xix. Common cause failure - is common mode a subset? For core protection calculators, were any software failures noted? How were these classified – common mode?	a. Current terminology is that common mode failures are a subset of common cause failures. b. No core protection calculator software faults were identified but if they had, they would have been included in the analysis as common cause failures.
68	SPSB:DSSA: NRR	(CE) Table 4 and Table 5 With respect to trip breaker configuration, do not seem to agree as to number and name (trip breakers or relays).	Modifications have been incorporated that provide consistency throughout the report, including numbers and names.
69	SPSB:DSSA: NRR	(CE) Although the CE report provides a summary of IPE/PRA RPS unavailability for comparison with the study results – did not find such a comparison for B&W.	Section 3.3 in both reports has been updated as necessary to provide consistency in the comparisons of the unavailability based on operational data with that of PRAs and other studies.
70	SPSB:DSSA: NRR	(CE) Page 36, Section 3.4. First paragraph states that earlier studies found RPS unavailability ranging from 1.5E-5 to 6.0E-5 and the CE reports obtained values from 1.5E-5 to 8.6E-6. The report states that the CE report found the values to be significantly lower?	The term "significantly" was used in error. The term "significantly" was changed to "slightly." The computed values for the CE RPS were lower but by less than an order of magnitude from the values reported in earlier studies.

ID	Commenting Organization	Specific Comment	Resolution Details
71	SPSB:DSSA: NRR	(CE) Page 45, Section 4.3.2. States that the unavailability from failures detected during operation and the unavailability from the failure modes detected during testing were calculated separately. Why? Doesn't this ignore instrumentation that may be failed but not detected until outage testing? (CE) Page A-13. Second Paragraph. Stated that operational data was used – why not both? (B&W) Page A-17. Second Paragraph. States that when differences were noted in operational data, and shutdown testing the operational testing data was used. The operational test data may limit the data for some components including transmitters. The use of operational testing may eliminate valid test results. (CE) Table C-3. It seems that by ignoring shutdown testing significant amount of data is being lost. (CE) Page C-16. Fifth Paragraph. CPR data during operations is used – transmitter tests at power only consist of channel checks and gross failure identification. Transmitters are tested at shutdown. It does not appear that this data should be deleted.	The decision to use operating data over shutdown data in the unavailability calculations was only made when statistical differences were noted between the operating and shutdown data sets. Otherwise, both sets were used. It was recognized that some components might have failed during operation but not be detected until shutdown. Likewise, some components might have failed during shutdown but not be detected until operation. The data failure reports were not sufficiently detailed to determine the exact failure time, therefore, the data was assigned to when it was detected. Likewise, It was recognized that there are some components, such as sensors/transmitters, that are tested more rigorously during shutdown than during operations. However, there was more confidence in the reportability of the number of demands during operations than at shutdown, making the operational data potentially less biased. Additionally, sensitivity analysis showed that there would be a negligible affect on the overall RPS unavailability even if the unavailability of the instrumentation such as the sensor/ transmitters was one or two orders of magnitude higher during shutdown than during operation because CCF of other components were more important that those of sensors.
72	SPSB:DSSA: NRR	(CE) Is the combining instrument data especially relay data with PWRs appropriate? The relay types for CE plants can be unique (MDR) with different failure mechanisms.	As shown in the "conclusion" column of Table C.1, the analysis of the instrumentation data identified that relay data for the CE study should be limited to only CE plants. Only mechanical and shunt trip breaker data were found to be poolable with other PWR vendors. The relay data was subsequently divided into logic relays and trip relays to recognize their unique characteristics and roles in the overall RPS designs. The limited amount of data on failures and associated demands prevented any further division of the relay data.

ID	Commenting Organization	Specific Comment	Resolution Details
73	SPSB:DSSA: NRR	(CE) Page A-4. Third Paragraph. Commercial *nuclear* power plant experience?	The sentence has been modified to clarify that the data classifiers do have commercial nuclear power plant experience.
74	SPSB:DSSA: NRR	(CE) Page A-5. First Paragraph. References monthly testing. Is this true for most data? Functional testing is generally on a quarterly basis. (CE) A-9, A-1.2.2.2. Surveillance tests are generally now quarterly and may have been quarterly for a significant amount of data. In addition, shutdown data could be 12, 18, or 24 months intervals.	The reference to monthly testing was in error. The frequency of surveillance tests used in the CE analysis has always been quarterly. Additionally, the computations for cyclic surveillance tests were based on the actual plant outage information provided in the monthly operating reports, this accurately reflects the shutdown intervals such as 12, 18 or 24 months. Appropriate text revisions have been made to clarify these points.
75	SPSB:DSSA: NRR	(CE) Page A-20 A-2.1.3.2. First Paragraph States that "…failure rate for Babcock and Wilcox plants…" Should this state CE?	The statement has been corrected to properly reference CE instead of B&W.
76	SPSB:DSSA: NRR	(CE) Page B-1. Components CPA and CPD are not in the Tables.	Tables have been updated to include all appropriate components (e.g., Table B-1 includes only components with degradation values of 0.1 or smaller.)
77	SPSB:DSSA: NRR	(CE) Page C-7. Report states that digital and analog core protection calculators showed no difference in reliability. Table C-2 and Tables C-3, C-5, and C-8 seem to show significant differences in failures recorded.	The tables in Appendix C show the results of the basic component failure probabilities that are used in the fault trees. These fault trees are used to determine the overall RPS unavailability. Even though there were some significant differences in the number of failures recorded for some components, the net result showed that there was little difference in the overall unavailabilities between the various CE RPS designs.
78	SPSB:DSSA: NRR	(CE) Page C-11. Flowchart" significance between plants" just among CE units?	The analysis steps shown in the flowchart are applied after the vendor evaluation is completed which determines whether data should be pooled. Therefore, the analysis may not be just among CE units.
79	SPSB:DSSA: NRR	(CE) Page C-33. Why a 90% confidence interval and not 95%?	The level of coverage for uncertainty bounds is a matter of choice. For consistency with most PRAs and other reliability studies, such as WASH-1400 and the previous RPS reports, we have selected the 90% confidence interval. If desired, the 95% confidence interval can be derived from the distribution data listed in Tables 3-1 and 3-2 of both reports.

ID	Commenting Organization	Specific Comment	Resolution Details
80	SPSB:DSSA: NRR	(CE) Page E-4. Do not understand failure criterion. Graphic confusing? One out of two twice criterion for 2 out of 4?	Appropriate sections of both reports have been revised to clarify that the channel logic is based on any 2-out-of-4 except for the reactor trip breaker portion which is based on 1-out-of-2/twice.
81	SPSB:DSSA: NRR	(CE) Page E-6, E-2.2.4. Are CE surveillance performed on a staggered basis? GE study is referenced. Do GE plants perform surveillance on a staggered basis?	a. Discussions with current resident inspectors confirmed that most CE plants perform surveillances on a staggered basis. b. The erroneous reference to the GE plants was corrected to read CE.
82	DET:RES	As we understood, there were no total systems failures in 231 demands for B&W plants and no total failures in 612 demands in C-E plants. A high number, 0.5, has been used in the report for introducing a failure or partial failure in accordance with "Jeffery's non-informative prior" as cited in the studies. We suggest that the reports should explain why such a high number was used as opposed to a number closer to zero.	As stated in the resolution to Comment 17, all discussions referencing the use of this computation technique to estimate the RPS unavailability have been removed from both reports. The RPS unavailability values presented in the reports are based strictly on using appropriate fault tree models.
83	DET:RES	Acronyms should be defined once and used consistently thereafter. The terms should be written out for the executive summary. See especially ac/dc vs. AC/DC.	Text has been modified as appropriate to improve consistency.
84	DET:RES	Terms like 5th, 95th, %, >, in the text (not equations) should be written out.	Text has been modified as appropriate to improve consistency.
85	DET:RES	Bibliographical endnotes should be numbered consecutively throughout the text.	Text has been modified as appropriate to improve consistency.
86	DET:RES	Section and subsection headings would look better left-justified throughout. Indents of paragraphs are unusual for NRC documents. Are they desirable? If so, spacing between paragraphs may be omitted. If paragraphs are not indented, then a consistent spacing practice should be observed.	Minimal changes have been incorporated since the current format is consistent with previously issued reports in this series.
87	DET:RES	Figure and chart titles should be the same font, bold or not, with initial caps throughout the report and appendices. The C-E report is much better in the text but some attention is needed. Figure and chart headings in many places should also be left-justified.	Figure and chart titles have been modified as necessary for consistency within each report and with previously issued reports in this series.
88	DET:RES	Latin abbreviations should be set off from the rest of the sentence in a consistent manner; either ; i.e., or (i.e., ...) but not both.	Text has been modified as appropriate to improve consistency.
89	DET:RES	Greek letters in the text should be in used consistently: α or alpha.	Text has been modified as appropriate to improve consistency.

ID	Commenting Organization	Specific Comment	Resolution Details
90	DET:RES	In the B&W report - "Terminology," The definition of "Diverse electronic trips" uses the term "diverse" in the definition. It should not. Suggest use of a thesaurus.	The definition has been revised to exclude use of the term being defined from the definition.
91	DET:RES	In both reports, Section 1, paragraph 2, the use of a Federal Register notice as a requirement citation may not be appropriate.	The reference to the Federal Register notice is per the NRC style guidance.
92	DET:RES	In the C-E report, much of Figure 1 is unreadable.	This figure is imported from a 3rd party source. It has been modified as best as possible to help its readability.
93	DET:RES	In the B&W report, table 2, Oconee trip modules - two-out-of-four configuration - is this 2/4 relays per module or 2/4 modules? Also define "gating power" where the term is first used.	Terminology has been clarified and gating power has been defined.
94	DET:RES	In the B&W report, Figures 3 and 4: SIMPLIFY - there are too many lines. Eliminate where ever possible and retain clarity. Ditto for labels. Labels should not cross lines. Clean up drawing.	Figures have been revised to clarify and simplify.
95	DET:RES	In the B&W report, Page 26 Paragraph 1 " (QT's)" Eliminate the apostrophe.	Use of the abbreviation has been eliminated from this paragraph.
96	DET:RES	In the B&W report, Figure 8, redraw. X-axis labels should be larger and less wordy. Use a legend if necessary. In the C-E report, the figure number is 11. It needs a similar fix. Also, put the footnote above the title in a smaller font.	Figures have been revised to simplify.
97	DET:RES	In both reports, footnotes in text should generally be in a font smaller than they are.	Font size has been decreased.
98	DET:RES	In both reports, page A-9: OUTINFO, do not italicize. B&W only: Davis Besse is misspelled.	Changes and corrections have been incorporated as requested.
99	DET:RES	In both reports, tables B1-B3 would be more easily understood with appropriate lines.	No changes have been made since this format is consistent with previous reports. Also, the tables are imported directly from other programs such that format changes are limited.
100	DET:RES	In both reports, Appendix C titles are verbose. In the C-E report, figure C-1 needs to go back to the drawing board, literally.	Changes have been made as best as possible. Fig. C-1 was directly imported and is the best available.
101	DET:RES	In both reports, Appendix D charts appear to be upside down.	No changes have been made since this is the common way to show event trees - from the top most event down to the lowest event.

ID	Commenting Organization	Specific Comment	Resolution Details
102	DET:RES	In both reports, Appendix D chart titles are not in the same configuration as other charts in the report. Numbering of the Appendix pages should be by machine in the C-E report.	Report changes have been made as best as possible. Some changes could not be accomplished since the charts, etc., were developed and imported directly from other programs, such as SAS and SAPHIRE
103	DET:RES	Since the reports will be printed in black and white, suggest the shading be eliminated for clarity. (Original was probably in color).	Shading has been eliminated for clarity.

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REMARKS

Final Reports: Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998, NUREG/CR-5500, Vol. 10 and Reliability Study: Babcock & Wilcox Reactor Protection System, 1984-1998, NUREG/CR-5500, Vol. 11

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