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Report on The Implications of the ATWS Events at the Salem Nuclear Power Plant on the NRC Program for Collection and Analysis of Operational Experience

Prepared by

Program Technology Branch

Office for Analysis and Evaluation
of Operational Data
U.S. Nuclear Regulatory Commission

The subject matter is under continuing review. This report supports ongoing AEOD and NRC activities (particularly the Salem Generic Implications Task Force) and does not represent the position or requirements of other NRC program offices.

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Executive Summary

In late March 1983, AEOD initiated a study to review and evaluate the implications of the ATWS events at Salem Unit 1 on February 22 and 25, 1983 on the NRC's program for collecting and analyzing operational experience. The study focused on two issues in particular. One issue concerned the adequacy of NRC's reporting requirements as they relate to reactor trip circuit breaker (RTB) failures, including the licensees' understanding of the requirements and the impact of proposed revisions to the requirements. The other issue concerned whether trends and patterns analyses of the reported RTB failures would have identified a significant potential for the problem at Salem Unit 1 before it occurred. Additional topics reviewed were the requirements for licensees to analyze operating experience with a specific focus on the ability to reconstruct the sequence of events.

The pertinent regulatory reporting requirements in Title 10 of the Code of Federal Regulations (10 CFR - Parts 21, 50.36, and 50.72), the "Reporting Requirements" section of the technical specifications for each pressurized water reactor (PWR), and Regulatory Guide 1.16 ("Reporting of Operating Information - Appendix A Technical Specifications") were reviewed, as were the Licensee Event Reports (LERs) submitted to the NRC. The Nuclear Plant Reliability Data System (NPRDS) was reviewed for relevant RTB engineering and failure data and was assessed for its future role as a component failure data base. In addition, the proposed revisions to 10 CFR 50.72 (Immediate Notification of Significant Events) and the proposed new 10 CFR 50.73 (Licensee Event Report System) were reviewed in relation to the requirements to report RTB or similar component failures.

The organizations involved in operating experience reviews and analysis and the results of the screening of reported RTB failures prior to the Salem

ATWS events were reviewed. The RTB failure data from events reported prior to 1983 is reviewed statistically; displayed in tabular and graphic form; grouped, transformed and analyzed for trends and patterns.

Finally, the requirements in 10 CFR 50 (Appendix A, "General Design Criteria ...", and Appendix B, "Quality Assurance Criteria ..."); regulatory guides; industry standards and NUREG documents applicable to operating experience analysis and event reconstruction were reviewed and the need for improvements assessed.

The following are conclusions from the above activities:

. Reporting Requirements (see Section 2.0)

(1) The regulatory reporting requirements are clear regarding the reporting of system level failures of the RPS, i.e., ATWS events. The prompt reporting requirements in license technical specifications and the existing rule for prompt notification (50.72) contain clear requirements in this area.

(2) The reporting requirements are less clear regarding the reporting of all individual RTB failures in operating plants. Close scrutiny of the license technical specifications and the associated Regulatory Guide 1.16 indicates that at least four criteria for reporting could apply. Thus AEOD's interpretation is that individual RTB failures occurring during operations or surveillance testing are reportable. Some failures, such as those identified during post-maintenance testing that were corrected prior to declaring the RTB operable and were a direct result of the maintenance activity, are not considered reportable unless they involved common cause, procedural inadequacies, or generic implications.

(3) Licensees generally understood the regulatory reporting requirements since 54 individual RTB failures were reported prior to the Salem events. However, there were some differences in interpretation of the regulatory reporting requirements and some RTB failures went unreported prior to the Salem ATWS events.

(4) The proposed changes to 10 CFR 50.72 and the proposed 10 CFR 50.73 will not require that all individual, random safety-related component failures, including RTB failures, be reported to the NRC. However, 50.73 will require reporting of multiple RTB failures, RTB failures that occur during a reportable event (e.g., a reactor trip), and RTB failures with a common cause or generic implications.

(5) When the proposed 10 CFR 50.72 and 50.73 are implemented, the NRC will rely on the NPRDS to collect individual component failure data, including RTB failures. Component failure data and related engineering data are an essential element of an effective operational experience program, and thus, a satisfactorily operating NPRDS is an integral part of the program. Although the data base did contain four RTB failures not reported in LERs, the existing NPRDS data on RTB failures was grossly incomplete when compared to the LER data. NPRDS did include some engineering data on circuit breakers in other plant systems of interest in the Salem review activities.

Operational Data Assessment (see Section 3.0)

(1) In 1971, 1979, and 1981, the review of LERs by the NRC identified specific concerns regarding the RTB and related UV trip device failures. As a result, these concerns were addressed by regulatory actions (the

issuance of IE Bulletin 71-2 on Westinghouse DB series RTBs, IE Bulletin 79-09 on General Electric AK series RTBs and IE Circular 81-12 on surveillance testing of RTBs). In addition, the equipment vendors also issued notifications regarding the proper care of RTBs (Westinghouse Technical Bulletin NSP-TB-74-1 and NSD Data Letter 74-2; General Electric Service Advice Letter 175 (CPDD) 9.3). Thus, specific safety problems had been identified with these components through the review of operational experience. However, the common cause failure potential and broader generic implications of the individual RTB failures were not fully appreciated.

(2) A review of the actions taken and in progress within the various offices of the NRC indicates that the NRC's operational assessment program was being implemented and a number of RTB problems had been recognized and studied. For example: (a) the need to test the UV trip attachment separately from the shunt attachment was identified by AEOD and formally issued as regulatory guidance by IE; (b) generic concerns with the GE type AK-2-25 breakers were recognized by Region V and formally documented; (c) specific problems with the preventive maintenance program for the RTB breakers at Salem were recognized by the resident inspector and formally documented in an inspection report; (d) the August 1982 RTB failure at Salem Unit 2 was recognized as a potential safety concern within NRR and discussed with NRR management and other NRC personnel during a periodic briefing; (e) the potential importance of the January 1983 RTB failure at Salem Unit 2 had been recognized by IE and to a lesser degree by AEOD and NRR, and was being actively investigated; and (f) a comprehensive study was made of RTB failures and their contribution to an ATWS event was assessed in 1982 by NRR in support of the ATWS Task Force activities. Thus, the various offices were involved and informed, but an overall program to address the safety concern was still in development (i.e., ATWS rulemaking).

(3) The Salem ATWS events again clearly emphasize that operational data assessment requires: clear and in-depth licensee reports; the ability to recall past failure histories with precision and completeness; the ability to analyze the failure history with regard to such aspects as the frequency, age, and location (specific plants involved); and the ability to identify the application of safety-related equipment which may be subject to the same types of failures. As a result, there is even a greater incentive for: the type of reports required by the proposed LER rule and by the NPRDS; improved and fully effective LER and NPRDS data bases that can be accessed and searched by all organizations involved with operational data assessment; a comprehensive and systematic trend and pattern analysis program; and a current and complete NPRDS engineering data file.

Trends and Patterns (see Section 3.4.2)

(1) Because of (a) the observed failure rate of RTBs, which was about as expected, (b) the apparent random nature and low frequency of RTB failures at specific plants, (c) the multiple modes of failure and operating difficulties experienced with these breakers, and (d) the great volume of failure data on safety-related breakers and RPS components, it is highly doubtful that a statistically based trends and patterns analysis program alone would have identified the RTB failures as "outliers" or a safety problem worthy of specific in-depth engineering investigation. Such failures are indeed a problem, but in view of the characteristics of this situation, the identification of this particular problem by the NRC would more likely be (and was to a degree in this case) a result of reviewing 10 CFR 50.72 and daily regional and LER reports from an engineering perspective. Trending of the breaker operational performance parameters, such as the time for breaker operation or tripping force, by licensees should, however, identify incipient failures.

(2) Trends and patterns analyses alone can only provide guidance on where to place engineering resources to further investigate "outliers" and irregularities in operating experience data. However, planned trends and patterns analyses will improve the routine display of collections of information, and when coupled with close scrutiny of failure data and detailed engineering assessment, particularly of those features related to reliability, can identify specific plant and/or generic safety problems and the need for corrective actions.

(3) Over the years, safety concerns for concurrent RTB failures in a redundant RTB system using only UV trip devices were tempered by the lack of a clear or persistent increasing trend of RTB failures and the fact that the overall failure rate was consistent with expected performance for circuit breakers. Actions taken on a plant-by-plant basis, and several times on a generic basis, were apparently sufficient to maintain an acceptable failure rate. However, followup actions were sometimes not as long-standing or as effective as planned. For example, at least one plant did not continue to test the UV device separately as requested in IE Circular 81-12; and previous guidance and lessons of experience, such as identified in IE Bulletin 71-2, seem to have been lost and forgotten with time.

(4) Based on calculations performed after the Salem ATWS events, the individual plant data do not cluster well around the NSSS averages. However, except in the case of Westinghouse, the statistical evidence is not strong enough to say that plants do not behave like members of an NSSS-defined population. In reality, the problems may still be a function of conditions at individual plants and, consequently, emphasis must be given to a close review of the failure rates and modes at individual plants. The calculated

estimates of overall rate of RTB failure based on reported RTB failures were within the uncertainty bounds of previous estimates, i.e., were essentially as predicted and thus acceptable.

(5) Even though the events at Salem involved no plant damage, no releases, and no immediate threat to public health and safety, the fact that the NRC and the industry have devoted extensive resources to studying its cause and implications is a strong indication of the heightened sensitivity to operational events and the progress made in understanding the lessons of operational experience. The Salem events could be considered precursors of very serious events that, it is hoped, will never happen because of the attention paid to the Salem events and the corrective actions that are being taken.

. Event Analysis and Data Collection (see Section 4.0)

(1) Operational event analysis and feedback by each licensee, the industry, and the NRC is essential for the safe operation of nuclear power plants. Several rules, regulations, and guidelines address this topic; however, they are not specific and, as a result, licensee activities vary widely. Although there are requirements that each licensee have procedures to carry out analysis and evaluation activities, each licensee makes an individual judgment regarding such items as what the elements of the program are; who is to be responsible for the program; to what degree independent evaluations are needed; what operational data is needed for the analysis program; what reviews are to be conducted and when; to what extent each event needs to be reconstructed; what constitutes a sufficient "understanding" of an event to allow resumption of plant operations; what weight should be given to long-term implications versus short-term plant

operations consideration; how the results of any analyses are to be fed back to operations personnel and incorporated into operational documentation; and how the involvement of management is increased with the seriousness or implication of events. Licensee programs to evaluate and analyse operational events, particularly reactor trips, need to be systematic and thorough to ensure available data are properly analyzed, significant failures are identified, the safety implications fully assessed, and proper corrective action completed.

(2) In order to perform an adequate post-event analysis and to properly identify anomalous behavior, events must be accurately reconstructed, including identifying initiating failures, causes for equipment operation or maloperation, activation of equipment alarms, operator actions involving equipment, and changes in plant parameters. A sequence of event recorder or combination of recorders is essential to provide the detailed information necessary to fully understand the cause, implications, and seriousness of an event. Personnel must also be trained to properly interpret and use the data and operational procedures must be developed to ensure proper implementation.

(3) The NRC and the industry have spent considerable time and effort defining the information needed by the operators to follow the course of a serious event <u>during</u> the event. However, the requirements for collection and storage of data to support <u>a posteriori</u> routine event reconstruction are not well defined and have been, at best, a peripheral issue in the assessment of the needs of the operators <u>during</u> the event. It is clear from past events (the Three Mile Island accident, the Salem Unit 1 ATWS events, and the Arkansas Nuclear One loss of offsite power event) that a systematic assessment of the unique needs of post-event reconstruction is needed. Such aspects as what information is to be recorded; scanning and recording rates; quantity of data recorded and retention period; and the

requirements for equipment availability, reliability, and qualification; need to be specifically addressed either by other organizations (such as INPO or ANSI) or by the NRC.

The following are AEOD's planned actions resulting from the study:

(1) Because of licensee interpretations of the reporting requirements and variation in the reporting of RTB failures, AEOD will draft proposed guidance to clarify the reporting of component failures such as RTB failures.

- Examples from RG 1.16 will be highlighted regarding the types of component and system failures to be reported.

- It will emphasize that reporting should be based on the nature of the event or failure and not on plant mode, or whether or not the system was in (or required to be in) an operable status.

(2) AEOD will continue to support efforts to assure: the routine and prompt reporting of component failures, particularly those leading to an LCO situation, to NPRDS; that NPRDS when properly implemented will meet NRC needs; and that utility participation in NPRDS is increased to acceptable levels. The progress toward these goals will be monitored and should it become clear that essential information will not be available from LERs or from NPRDS, AEOD will document the concern to the Commission noting possible options and providing a staff recommendation.

(3) Although this study indicates that the specific nature and characteristics of the reactor trip breaker problems make it unlikely that trends and patterns analysis of available data would have forecasted the Salem ATWS events, AEOD still believes that trends and patterns analysis is an important and integral part of the program for analyzing and evaluating operational

experience. Therefore, AEOD will continue the development of a statistical trends and patterns analysis program. Thus, trends and patterns in conjunction with engineering assessments of individual LERs will provide additional perspective and insight (both plant-specific and industry-wide) of LER-based operational experience data.

The AEOD trends and patterns program will include the statistical analysis and subsequent evaluation of the rate of occurrence of various types of events or occurrences. The analysis includes comparison of these rates among similar plants.

AEOD's trends and patterns program will not, however, include the routine collection and analysis over time of plant parameters (e.g., temperatures, currents, torques) associated with predicting degradation of components (e.g., data from surveillance testing or operational events). The feasibility, costs, and benefits of the routine collection and analysis of such data are currently being studied as part of the development of a more diversified data collection system for operational experience. It is currently estimated that the development of this more diversified data collection program will take at least three to five years to complete.

(4) In order to assure resolution of potential generic safety concerns, AEOD will review the handling of potential generic issues that are identified as a result of operational experience, to determine the clarity and adequacy of current NRC procedures and recommend improvements, if needed.

(5) AEOD will review and revise as necessary the AEOD screening procedure and associated tracking system in order to confirm that potential generic failures and events, particularly those with common cause failure implications, are receiving careful review, and to assure that proper documentation of such reviews and their associated disposition are entered in and easily retrievable from a computerized tracking system.

AEOD also endorses the following planned actions of other offices:

(1) NRR's commitment to review the current situation regarding post-event data acquisition and evaluation. This study again emphasizes the importance and role that plant computers play in event reconstruction, yet neither the NRC nor the industry have provided specific guidance or requirements for this essential activity.

(2) IE's commitment to increase emphasis on bulletin and circular follow-up and close-out and to assure associated licensing, inspection and other regulatory actions are promptly completed. The need for these activities is supported by this study.

1.0 Background and Statement of Intent

1.1 Background

On February 25, 1983, Salem Unit 1, a Westinghouse designed nuclear power plant, experienced a total failure of the reactor trip system (RTS) to automatically shut down the reactor upon receipt of a valid signal from the reactor protection system (RPS).* A similar event had occurred at Salem Unit 1 on February 22, 1983. The failures were caused by two electro-mechanical circuit breakers, referred to as reactor trip breakers (RTBs), failing to open in response to the automatic trip signal from the RPS. The RTBs did not trip open because the associated undervoltage (UV) trip attachment did not actuate the trip mechanisms. The breakers subsequently opened when the operators actuated them via the manual scram switch. The proper functioning of the automatic feature of the RTS including the RTBs is fundamental and of prime importance to the protection of public health and safety; its failure results in total reliance on operator actions to control plant transients.

With few exceptions, all three PWR NSSSs (Westinghouse, Babcock and Wilcox, and Combustion Engineering) use an RTS design requiring circuit breakers (either ac or ac and dc) to open to trip the reactor. Although the basic designs of the RTSs differ considerably among PWRs, they generally include a UV trip attachment and a shunt trip attachment to actuate the circuit breaker. The UV device initiates a breaker trip when de-energized, while the shunt device is energized to trip the breaker. The basic design schemes are as indicated in Table 1.1.

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* A detailed discussion of the technical and operational aspects of the events is contained in NUREG-0977 (published in March 1983), and the results of the Salem Task Group Review of the events and related issues and generic implications are contained in NUREG-1000 (published in April 1983). Therefore, this study includes technical details only as needed to clarify the text.

Table 1.1

Reactor Trip Circuit Design Schemes

NSSS	No. of Reactor Trip Breakers in Trip Circuitry	Breaker Actuating Devices for Trip: Automatic	Breaker Actuating Devices for Trip: Manual
Westinghouse 1/	2	UV	UV and shunt
Babcock & Wilcox 2/	6	UV	UV
Combustion Engineering 3/	8	UV and shunt	UV and shunt

1/ The Haddam Neck Plant uses the UV and the shunt device to trip the RTBs for automatic and manual trips; Yankee Rowe uses only shunt devices to trip the RTBs. The RTBs in use are Westinghouse DB or DS series breakers in most plants.

2/ All Babcock and Wilcox designs, except Davis Besse, use two ac circuit breakers and four dc circuit breakers. Davis Besse uses four ac circuit breakers. The RTBs in use are General Electric AK-2 series breakers.

3/ Palisades and Fort Calhoun use electrical contactors, rather than circuit breakers. The plants using RTBs have General Electric AK-2 series breakers.

Past RTB failures at pressurized water reactors (PWRs) other than Salem appear to have resulted from failure of the UV trip attachments to open the circuit breaker as discussed. There has been a long history of such failures. As a result, starting in 1971, a number of notifications of RTB and UV device failures and recommended corrective actions (such as IE Bulletins, and IE Circulars, and technical service bulletins) have been issued by the Atomic Energy Commission (AEC)/Nuclear Regulatory Commission (NRC), the breaker manufacturers, and the nuclear steam system suppliers (NSSSs) involved (see Appendix E).

A full understanding of the number and types of RTB failures is complicated by a number of factors, many of which have changed over the years. As a result, a relevant and accurate history of UV trip device performance and related RTB failures is difficult to reconstruct. For example:

- reporting requirements in terms of the type of event that must be reported have undergone major changes over the years;
- reporting requirements in terms of the equipment and conditions subject to reporting have also undergone changes over the years, particularly regarding equipment classification and limiting conditions for operation;
- surveillance testing of RTBs has varied over the years from plant to plant, and within specific plants with regard to frequency and completeness. For example, some test methods only verify the capability of the RTB to open, while others test the independent functioning of the UV device and the shunt device to open the RTB.

These variables, together with the differences in RTS design, have resulted in inconsistencies and perhaps in incompleteness in the identification of RTB failures and associated data.

1.2 Statement of Intent

This AEOD study was initiated to review and evaluate the implications of the Salem events on the NRC's program for collecting and analyzing operational experience. The study addresses the adequacy of NRC's current reporting requirements and the licensee's understanding of them as they relate to RTB failures. In addition, the study reviews whether there was a significant potential for simultaneous failure of the RTBs that the existing industry

and NRC operating experience assessment programs should have recognized. A specific focus of the study is to develop the facts, discuss implications, and provide relevant observations on two issues: (1) Are existing event and failure reporting systems adequate and generally well understood by licensees? and (2) Should trends and patterns analyses have identified the problem at Salem before it occurred? Associated topics also covered include data availability, the requirements for licensees to perform analysis of operational experience, and post-event reconstruction activities. The impact of planned revisions to the reporting requirements is also addressed for the data available prior to the Salem events.

This study, and specifically the data analysis sections, is generally limited to the RTBs in PWRs, but there is some discussion on similar circuit breakers in other nuclear plant systems.

2.0 Reporting Requirements and Reports Received

2.1 Regulatory Reporting

Each licensed nuclear power facility must report certain types of operational events to the NRC. Reporting requirements are delineated in various parts of Title 10 ("Energy") Code of Federal Regulations (10 CFR) Chapter 1 and in each licensee's technical specifications and/or license provisions. Those of specific interest are in 10 CFR 50.36. Regulatory reporting requirements have existed since the beginning of licensing activities, and initially were stated in the individual licenses for nuclear power plants. The regulatory intent has always been to have items of potentially serious safety concern reported, such as the failure of protective devices to function properly.

In the early 1970s, Regulatory Guide 1.16, "Reporting of Operating Information," was developed to provide uniform guidance on the information to be supplied to the regulatory body (the Atomic Energy Commission at the time). During the early to mid 1970s, the importance of the collection, assessment and feedback of operational experience grew as the nuclear power industry expanded. As a result, Regulatory Guide (RG) 1.16 was revised several times to clarify what information was desired. Revision 3 to RG 1.16, issued in January 1975, not only specified the types of events to report, but also contained many notes and examples for added guidance. Revision 4 to RG 1.16 was issued for public comment in August 1975; however, the sections pertaining to the type of events to report, and the associated notes and examples, had not been changed. As discussed further in Section 2.1.1, all plant technical specifications presently contain reporting requirements based on (and are essentially identical to) the reporting criteria in RG 1.16.

2.1.1 License Technical Specification Requirements

10 CFR 50.36, "Technical Specifications," specifies items to be included in individual plant technical specifications, including specifying requirements for the licensee to notify the Commission when safety limits are exceeded, or limiting conditions for operation are not met. In addition, the technical specifications also include requirements, titled "Reporting Requirements," specifying when each licensee must submit certain reports to the Commission. Included in these requirements is the reporting of certain off-normal events ("Reportable Occurrences") involving safety-related matters including component, system, and structure failures. The technical specifications define two types of Reportable Occurrences: (a) those requiring prompt notification with

written followup within two weeks, and (b) those requiring only thirty-day written reports. The written reports, whether two-week or thirty-day, are known as Licensee Event Reports (LERs).

In October 1974, NRC's Office of Nuclear Reactor Regulation (NRR) initiated a program to update the reporting requirements specified in the technical specifications of power reactor licensees. Licensees were issued specifications that either referred to RG 1.16 or used the criteria from the Guide.

Based on the later revisions to RG 1.16, NRR initiated action in 1977 to again update the reporting requirements in the technical specifications. As a result, all plant technical specifications now contain reporting criteria essentially identical to RG 1.16 regarding the types of events to report. However, none contain the specific examples of RG 1.16. In addition, only some plant technical specifications include the clarifying notes from RG 1.16. A typical set of reporting requirements which do contain the clarifying notes is shown in Appendix A.

Experience has shown, however, that the benefits of standardizing these reporting requirements have not been fully realized, not only because of differing interpretations of these requirements, but also because other sections of the technical specifications (e.g., LCOs - limiting conditions for operation), to which the reporting requirements refer, remained non-standardized among plants. Thus, even though all plants must report when they enter an LCO, the LCOs vary among plants and as a result the reporting varies.

Furthermore, the technical specifications seem to be silent on whether or not failures are reportable which are found during plant modes when equipment is not required to be operational (i.e., when the system is not subject to an LCO).

For purposes of this study, RTB failures are defined as not only failure to operate, but also as sluggish or delayed operation. The latter, examples of which have ranged from seconds to minutes, are also significant in terms of anticipated transients without scram (ATWS) events.

The guidance in RG 1.16 and typical technical specifications have been interpreted within AEOD to require reporting of RTS and RTB related failures under the following conditions:

(1) Failures of the reactor to trip (either automatically or manually) are clearly reportable (see Appendix A, Paragraph 1.2.a.1 of this report). The reporting requirements are very specific regarding total system failure to function. Such an event is also reportable under 10 CFR 50.72, which became a regulatory requirement in February 1980 (see Section 2.1.2). The February 22 and 25 failures of Salem Unit 1 are the only known cases of a PWR failure to automatically trip upon receipt of a valid RPS signal.

(2) A failure of the UV trip device, or any other failure which causes one RTB to fail to open on an operational demand or a surveillance test demand is considered reportable under section 1.2.a.9 of Appendix A based upon the potential generic implications of the failure. The current version of RG 1.16 contains an example under this reporting criterion that is analogous to many RTB failures due to UV trip device failures. The example reads, "Failure of magnetic trip mechanisms on a safety-related circuit breaker to provide trip on instantaneous overcurrent as indicated on the manufacturer's time-current characteristic curve."

(3) A failure of the RTB to open during surveillance testing is considered reportable under Appendix A, Paragraph 1.2.b.2 for plants that have limiting conditions for operation imposed on the reactor trip breakers. For those plants without LCOs on the RTBs, reporting could be required under 1.2.a.9, 1.2.b.1 or 1.2.b.3. The intent of the Regulatory Guide seems clear in this regard, and is properly interpreted and implemented by most licensees. For example, the current version of RG 1.16 contains an example for Appendix A, Paragraph 1.2.b.1, which though portraying an event involving a UV relay (rather than a UV trip attachment) that does not actuate at the proper setpoint to trip the RTB, provides for the reporting of a single random failure in the RPS. The example reads, "During test, one out of four undervoltage relays failed to perform its function of tripping a reactor trip breaker."

Some of the events of interest in this study involved failures of the UV trip device to trip the RTB during surveillance tests, even though the shunt trip device operated properly. The reportability of failures of the UV device to actuate the RTB is implied, though not specifically stated even when the shunt devices are operable, since the UV device is credited as safety related due to its fail-safe feature on loss of power.

Finally, the regulatory interest in circuit breaker failures, particularly RTB failures, evidenced by the issuance of Bulletins and Circulars over the years, also should have influenced licensees to interpret the need for reportability.

In summary, current reporting requirements indicate that failure of the RTS to trip the reactor is clearly reportable, and the random single failure of an RTB to open in response to an actual or test signal is also considered reportable by AEOD under operating or surveillance test conditions. Reporting of failures associated with testing after maintenance, before operability of the breaker is required, is not explicitly required by existing technical specifications.

2.1.1.1 Reports of RTS Failures

The Salem Unit 1 events were the only reported RTS failures while a PWR was operating. However, prior to the February 22 and 25, 1983 RTS failures at Salem Unit 1, there was one event where both redundant reactor trip breakers in an RTS did not open (during testing while shut down) due to concurrent failure of the UV devices. It occurred during a special test of the UV tripping capability at Haddam Neck on December 2, 1971 and was reported as event No. 71-1 (see Appendix B). The testing was initiated as a result of malfunctions of UV trip devices at another facility. When both redundant shunt trip devices were inspected at Haddam Neck (during a plant shutdown), they were found to operate normally. Then, fuses were removed from the shunt trip circuit and a trip signal was sent only to the two UV tripping mechanisms. Neither mechanism tripped. Inspection showed that there was excessive friction between the trip lever and the UV device, apparently caused by a combination of surface contamination and rough surfaces.

Even though the RTBs at Haddam Neck have always opened on an operational demand, this event identified a concern, as early as 1971, that multiple concurrent RTB failures due to concurrent failure of the UV devices can occur. Therefore, in this regard, it is a precursor to the Salem Unit 1 events.

The Haddam Neck event, together with the three independent failures during 1971 at Robinson Unit 2 (see Appendix B), were the basis for IE Bulletin No. 71-2 issued on December 9, 1971 which informed licensees of the regulatory concern for the RTB failures (see Appendix E).

2.1.1.2 Reports of RTB Failures

In order to retrieve the RTB failure reports from the event reports received between 1971 and February 1983, searches were made by AEOD of available computerized data bases (the NRC and Oak Ridge National Laboratory's (ORNL) Nuclear Operations Analysis Center LER files, and the Institute of Nuclear Power Operation's (INPO's) Nuclear Plant Reliability Data System (NPRDS) file). The results are summarized in Appendix B. This tabulation identifies a total of 43 licensee reports representing a total of 54 RTB failures reported prior to the Salem Unit 1 ATWS events. In addition, four RTB failures submitted as NPRDS reports only are also listed. Failures not previously reported to the NRC, but which recently came to the NRC's attention as a result of the Salem investigations, are not included in Appendix B; they are addressed separately in Section 2.1.4. Where the licensee has given the basis for reporting, it also is listed. The licensees' bases for reporting are generally contained in their cover letters forwarding LERs to the NRC. The range of variability in the interpretation of the criteria for RTB failures is shown in Appendix C. The variations shown in Appendix C, together with the fact that some licensees had not reported some RTB failures (see Section 2.1.4), may be indicative of a combination of lack of specificity in the reporting requirements and inadequate understanding by some licensees of the intent of the requirements.

2.1.2 10 CFR 50.72 - Notification of Significant Events

2.1.2.1 Reporting Requirements

After the Three Mile Island accident, a regulation (10 CFR 50.72) was implemented in February 1980 as an immediately effective rule. The intent of this regulation was to improve the NRC response to abnormal events in nuclear power plants by requiring licensees to notify the NRC Operations Center no later than one hour after certain specified significant events. The requirements of 10 CFR 50.72 are shown in Appendix D. As currently written, all reactor trips (whether successful or not) are reportable. If a single RTB failure occurred during a successful reactor trip, the failure (if known at the time) would be reported as part of the description of the reactor trip. The failure of a single RTB (either on demand from the RPS or during surveillance testing), however, is not a specific reason for reporting under 10 CFR 50.72.

2.1.2.2 Reports Received

Both of the February 1983 Salem Unit 1 events were reported to the NRC Operations Center. The February 22, 1983 event was initially reported as a manual reactor trip because the failure of the RTS automatic trip was not recognized at the time. (On February 23, the licensee indicated to NRC personnel that an automatic trip had preceded the manual trip. This subsequently was determined to be inaccurate.) The February 25, 1983 event was reported as an RTS automatic trip failure with a subsequent manual trip.

The January 6, 1983 event at Salem Unit 2 was also reported to the NRC Operations Center as a reactor trip. The fact that an RTB hung-up for 25 minutes was not

part of the immediate report (probably because it was not known immediately). However, this failure was included in the subsequent daily report from the NRC Regional Office.

2.1.3 10 CFR 21 - Reporting of Defects and Noncompliance

2.1.3.1 Reporting Requirements

During 1977, 10 CFR 21 became effective. This regulation requires that a director or responsible officer of a firm constructing, owning, operating or supplying the components of any licensed facility report information reasonably indicating: (a) that the facility, activity, or "basic component" supplied to such facility or activity fails to comply with the Atomic Energy Act of 1954, as amended, or any applicable rule, regulation, order, or license of the Commission relating to substantial safety hazards; or (b) that the facility, activity, or "basic component" supplied to such facility or activity contains defects which could create a substantial safety hazard; he is to immediately notify the Commission of such failure to comply or of such defect, unless he has actual knowledge that the Commission has been adequately informed of such defect or failure to comply [emphasis added]. As used in this regulation, "basic component" when applied to nuclear power reactors means a plant structure, system, component or part thereof necessary to assure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10 CFR 100.11.

Thus, the coverage and intent of 10 CFR 21 are clearly directed to the identification of real and potential problems with critical components such as RTBs.

2.1.3.2 Reports Received

IE's log of Part 21 reports received since 10 CFR Part 21 became effective was reviewed. No reports pertaining to the Westinghouse DB type or General Electric AK type RTBs were located prior to the Salem Unit 1 ATWS events. This may be due to the fact that a report would not be required if the Commission has been adequately informed of a defect or failure to comply. The relevant history in Appendix E provides evidence that the NRC was made aware of individual problems with the RTBs.

Subsequent to the Salem Unit 1 ATWS events, Westinghouse Product Division on April 20, 1983 filed by telephone a 10 CFR Part 21 report concerning misoperation of DS-416 RTB with UV trip devices. The potential deficiencies involve clearance and dimensional problems and retaining ring seating defects in the UV device which could impede proper tripping of the associated RTB. Five operating units and 24 units under construction could be affected.

2.1.4 Other Failures of RTBs

There have been other instances of RTB failures, which occurred prior to the Salem Unit 1 events in February 1983, which were not reported to the NRC by LERs. The events, together with the known reasons for not reporting as LERs, are discussed below.

2.1.4.1 Failures Reported by Information Letters

The licensee for Robinson Unit 2 reported two events, occurring on September 23, 1981 and December 20, 1982, to the NRC in the form of information letters rather than LERs. Following the first event, the NRC Resident Inspector

felt that an LER should be submitted but the licensee stated that their existing technical specifications do not address RTB failure as being either a limiting condition for operation or a reportable occurrence. Thus, the licensee believed an LER was not required, but did agree to provide an information report to the NRC. An information letter also was submitted for the second event. The use of information reports is not a desirable substitute to the formalized and computerized LER systems since it may lead to important failure data being unavailable to those in the NRC and industry who analyze such information.

2.1.4.2 Failures Reported to NPRDS

As shown in Section 2.2.2 below, only seven RTB failures, from six different units, were reported to the voluntary NPRDS. Of these seven failures, four were not reported to the NRC as LERs. The reasons why these four were not reported as LERs are unknown.

2.1.4.3 Failures Reported Associated With IE Bulletin 83-04

In March 1983, (i.e., after the events at Salem) the licensee for San Onofre Units 2 and 3 tested their General Electric AK type RTBs (though not required to do so at the time). During the testing, several failures of the UV trip attachment were experienced. During the investigation of these events, it was learned that similar failures had occurred in March and July 1982, which had not been reported to the NRC.

Based on the March 1983 failures at San Onofre Units 2 and 3, previous failures of the General Electric AK type breakers reported by other plants, and the discovery of previously unreported events at San Onofre 2 and 3, the NRC issued

IE Bulletin No. 83-04 to all PWR licensees. For licensees with other than Westinghouse DB type breakers, the Bulletin included requirements for (1) RTB testing, (2) reporting results to the NRC, and (3) providing a description of any RTB malfunctions not previously reported to the NRC.* As of April 1983, five licensees have described previously unreported RTB malfunctions. Appendix F summarizes these events. Most licensees provided reasons why the failures had not been reported (see Appendix F), even though the Bulletin did not specifically request such information. The cognizant Regional Offices and Resident Inspectors are evaluating the reportability requirements for these events.

Primary reasons given for not reporting were that the failures occurred prior to issuance of an operating license (e.g., during pre-operational testing) or while the RTBs were not required to be in an operable state (e.g., while the plant was shutdown, or while the system was considered inoperable).

In addition to the previously unreported RTB failures, Appendix F also includes RTB failures which have occurred after the February 1983 Salem Unit 1 events.

2.2 Nuclear Plant Reliability Data System (NPRDS)

2.2.1 Reporting Requirements

NPRDS is a computerized data base which is intended to contain engineering data and failure data on Safety Class 1, 2, and 1E components and systems (approximately 3,000 to 4,000 items per plant) at participating operating nuclear power plants. The data base is designed to be used to identify

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* IE Bulletin 83-01, issued to licensees using Westinghouse DB-type breakers, did not include a requirement to report previously unreported breaker failures. Actions are currently in progress to obtain information on any previously unreported RTB malfunctions from licensees with Westinghouse DB type breakers.

applications of specific components and to produce failure data on such components and systems. Development of the system began in the spring of 1973 under the direction of the American National Standards Institute (ANSI) Subcommittee N18-20, which included representatives from the utilities, nuclear steam system suppliers, and the AEC. Licensee participation has always been on a voluntary basis; unfortunately, due to the limited participation by nuclear power plant owners, the data base is incomplete and not current.

In January 1982, direction of the system was assumed by the Institute of Nuclear Power Operations (INPO) in an attempt to improve licensee reporting to NPRDS. Since NPRDS provides for the reporting of engineering data and failure data for Safety Class 1E components (defined in IEEE Standard 380-1975) which include the RTBs, engineering data and reports of failures of RTBs should have been reported to NPRDS by participating licensees.

The UV trip device, however, is not listed as a separate component and may have been considered as a piece-part of the RTB. Consequently, failures of the UV trip device would normally be reported to NPRDS only if the failure of the UV trip device caused a failure of the breaker to open.

2.2.2 Reports Received

In researching the NPRDS failure data for RTBs which failed to trip, seven failures were found from six different nuclear units (Table 2.1). This compares to the 54 failures from 20 different nuclear units (Appendix B) that were discovered from LERs and other reported information. Of these seven failures discovered in NPRDS, three were also reported to the NRC by LERs (i.e., about 6% of the failures reported as LERs were also reported to the NPRDS data base).

The NPRDS engineering data is the only industry-wide system for determining the use or application of specific components at nuclear plants (i.e., engineering data). It should be able to identify which plants have certain types of RTBs and where else in the plants similar breakers are used in safety systems.

Table 2.1
NPRDS Data on Reactor Trip Breaker Failures to Trip

Plant	Docket 50-	LER No.	Date of Event	Cause of Failure	Breaker Vendor	Breaker Model No.
1) Calvert Cliffs 1	317		2/28/78	Out of adjustment.	GE	AK-2A-25-1
2) Calvert Cliffs 1	317		5/3/79	Defective UV device.	GE	AK-2A-25-1
3) Calvert Cliffs 2	318		1/14/82	Defective UV relay.	GE	AK-2A-25-1
4) Davis Besse 1	346	81-70	10/26/81	Not specified.	GE	AK-2A-25-1KL
5) Kewaunee	305		3/25/76	Binding on UV coil linkage.	W	DB
6) Point Beach 1	266	76-11	11/30/76	UV device sticky.	W	DB-50
7) Zion 1	295	76-53	9/17/76	Dirty breaker.	W	DB-50

However, since the UV trip device was not specifically defined within the NPRDS scope, it is not possible to search for engineering or failure data directly associated with it or to specifically identify breakers using such a device.

A query was run on the NPRDS data base to identify the location of Westinghouse DB-type circuit breakers. Records for 300 Westinghouse DB-type circuit breakers located in various systems at 15 plants were found. This covered a range of three breakers per plant to 70 breakers per plant, with more than 50% of the plants reporting less than ten breakers each. A similar search for all GE AK-type circuit breakers used in PWR plants identified 198 circuit breakers in various systems at 15 plants. Unfortunately, it was frequently impossible to identify which of these breakers are the RTBs, because RTBs are coded as being in various systems including the RPS and the reactivity control system.

Finally, only about 60% of the PWRs provided any data on electrical breakers. Thus, for the remaining 40% of the plants, there was no data on any breakers.

2.3 Proposed Changes to the Regulatory Reporting Requirements

A proposed final rule has been sent to the Commission for approval to revise the reporting requirements for operational experience at nuclear power plants by establishing a new Licensee Event Report (LER) System. The proposed rule (10 CFR 50.73) would codify this LER system in order to establish a single set of requirements that apply to all operating nuclear power plants; it will define (1) the events and situations that must be reported, and (2) the information that must be provided in each report. The rule will reduce the number of reports to the NRC to those of greater potential safety significance, but require an expanded report content needed for careful study of events and conditions that might lead to serious accidents.

The rule is intended to overcome shortcomings in the present reporting requirements. The criteria for reporting have been selected to require the reporting of potentially significant events, including associated component failures,

regardless of the plant operating mode or power level, or the safety classification of the components, systems, and structures involved. Engineering judgment will still be involved to some degree on several of the reporting criteria, but actions to reduce the variability in reporting will include a supporting document providing detailed examples and formal workshops for NRC inspectors and licensees. Finally, the rule notes that assurance of safe operation of all plants depends on the accurate and complete reporting by each licensee of all events having potential safety significance, and it encourages the reporting of events having such potential significance even though they may not be directly required.

Consistent with the Commission direction to minimize the degree of overlap between LER and NPRDS reporting, most random single component failures will be reported to the NPRDS, instead of to the LER system. Thus, under the proposed LER rule, single RTB and similar component failures will only be reportable as LERs if the failure involved a common cause or generic implications, although random single failures occurring in conjunction with a reportable event would be identified and included in the LER.

Obviously, the Commission fully recognizes the necessity of NPRDS (see 46 FR 49134) and has indicated that reporting requirements are only being relaxed with the expectation that full utility participation, cooperation, and support of the NPRDS will be forthcoming under INPO's direction. If the NPRDS does not become operational at a satisfactory level, the Commission has indicated an intent to take the necessary remedial action in the form of additional rulemaking. Additional background material on 10 CFR 50.73 rulemaking is available in 46 FR 6793, dated January 30, 1980, SECY 81-494, and SECY 82-3.

2.4 Conclusions

(1) The regulatory reporting requirements are clear regarding the reporting of system level failures of the RPS, i.e., ATWS events. The prompt reporting requirements in license technical specifications and the existing rule for prompt notification (50.72) contain clear requirements in this area.

(2) The reporting requirements are less clear regarding the reporting of all individual RTB failures in operating plants. Close scrutiny of the license technical specifications and the associated Regulatory Guide 1.16 indicates that at least four criteria for reporting could apply. Thus AEOD's interpretation is that individual RTB failures occurring during operations or surveillance testing are reportable. Some failures, such as those identified during post-maintenance testing that were corrected prior to declaring the RTB operable and were a direct result of the maintenance activity, are not considered reportable unless they involved common cause, procedural inadequacies, or generic implications.

(3) Licensees generally understood the regulatory reporting requirements since 54 individual RTB failures were reported prior to the Salem events. However, there were some differences in interpretation of the regulatory reporting requirements and some RTB failures went unreported prior to the Salem ATWS events.

(4) The proposed changes to 10 CFR 50.72 and the proposed 10 CFR 50.73 will not require that all individual, random safety-related component failures, including RTB failures, be reported to the NRC. However, 50.73 will require reporting of multiple RTB failures, RTB failures that occur

during a reportable event (e.g., a reactor trip), and RTB failures with a common cause or generic implications.

(5) When the proposed 10 CFR 50.72 and 50.73 are implemented, the NRC will rely on the NPRDS to collect individual component failure data, including RTB failures. Component failure data and related engineering data are an essential element of an effective operational experience data program and thus, a satisfactorily operating NPRDS is an integral part of the program. Although the data base did contain four RTB failures not reported in LERs, the existing NPRDS data on RTB failures was grossly incomplete when compared to the LER data. NPRDS did include some engineering data on circuit breakers in other plant systems of interest in the Salem review activities.

3.0 Operational Experience Assessments

In July 1979, after the Three Mile Island accident, the Commission acted to make significant improvements in the way the NRC assessed the large volume of accumulating operational experience data (about 14,500 LERs at the end of 1978). An agency-wide program was approved which included the formation of a new staff office, the Office for Analysis and Evaluation of Operational Data (AEOD) that reports to the Executive Director for Operations (EDO). At the same time, the Commission specified that the individual program offices (NRR, NMSS, RES, and IE) have operational data analysis and evaluation capabilities.

During the same period, the NRC licensees and the nuclear industry (NSSSs, AEs, component manufacturers, and others) also recognized the importance of increasing the emphasis on operational data assessment. Varying degrees of commitment were made by the organizations involved to improve on the collection, analysis, and feedback of operational data. The utilities created two new

organizations - the independent Institute of Nuclear Power Operations and the Nuclear Safety Analysis Center within the Electric Power Research Institute. Individual licensees upgraded their operational data assessment programs to varying degrees, ranging from a minimum to meet newly imposed NRC requirements covering the assessment of operating experience with their plants and similar facilities to sophisticated programs to better feed back the lessons of experience to increase the safety and reliability of plant operation. The other entities (NSSSs, AEs, component manufacturers, and others) also revised their programs to the extent they deemed necessary to be more responsive to operating experience.

The following discussion summarizes the various entities and their activities in regard to the assessment of events and/or licensee event reports (LERs) received since 1980. The discussion focuses on events involving RTB and UV device failures or malfunctions in order to assess the implications from the Salem Unit 1 events on the operational data assessment activities.

3.1 Organizations Involved and Their Activities

The organizations currently involved and their activities related to the collection and analysis of nuclear power plant operational data, the initiation of corrective actions, and the dissemination of the lessons learned from experience are:

(1) Each NRC licensee performs the primary activities related to reviewing and analyzing its own operating experience and factoring that, plus the pertinent experience of others, into the safety of its operations. The general requirements and guidance for fulfilling these responsibilities are given in NUREG-0737 (Item I.C.5) and in Regulatory Guide 1.33, which endorses ANSI Standard 18.7 (ANS 3.2). The licensees' activities are paramount to

the success of the program; i.e., thorough evaluations must be conducted including the use of the operating history available from the plant, from similar plants, from vendors and from LER and NPRDS data bases available through INPO and the NRC.

(2) The industry (NSSSs, AEs, etc.) must assess the adequacy of their products and services and must notify the NRC and others regarding "significant defects and noncompliance." This reporting is required by law (Section 208 of the Energy Reorganization Act of 1974) and is reflected in 10 CFR 21 (see Section 2.1.3).

(3) Industry organizations, particularly INPO, conduct event analyses and generic evaluations on an industry-wide basis in order to identify potentially serious safety problems. Licensees have been given credit in their operational data assessment programs for the participation in INPO's SEE-IN program as a method of meeting the requirement to review the operational experience from other nuclear power plants. Thus, licensees who do not independently review the operating experience from other facilities have to determine the applicability of output from the INPO SEE-IN program to their facility, and to initiate appropriate corrective actions.

(4) NRC conducts oversight activities including activities to collect, analyze, and feed back operational data as part of its program to protect the public health and safety. Within the agency, these activities are distributed among several headquarters offices (AEOD, NMSS, NRR, IE, and RES) and the five regional offices. The NRC's operational experience program is defined in detail in NRC Manual Chapter 0515, "Operational Safety Data Review." A synopsis of each office's activities follows:

(a) AEOD's activities involve the analysis and evaluation of operational safety data associated with NRC-licensed activities, and the feedback of such analyses to improve safety. The office provides agency coordination of operational data collection, storage, and retrieval activities; independently analyzes and evaluates operational experience (e.g., LERs); coordinates abnormal occurrence determinations and issues the associated reports; feeds back the lessons inherent in operational experience to NRC licensing, standards, and inspection activities and to licensees; assesses overall effectiveness of the agency-wide safety data program; and acts as a focal point for interaction with outside organizations dedicated to operational safety data analysis and evaluation. AEOD's activities are generally based on LERs rather than an immediate response to operating events, though in the past, AEOD has provided support to selected immediate response actions.

(b) NRR's activities include participation in immediate response activities such as: prompt assessment of the safety significance of operational events (individual events and generic concerns) and the implementation of regulatory actions (orders, license modifications, generic letters, etc.) on a short-term as well as a longer-term basis; analysis and evaluation of operational data to assure the safety concerns are within the bounds of the Safety Evaluation Report and that existing safety margins are sufficient to ensure the public health and safety; providing feedback to individual licensees and industry organizations; and working with the other NRC offices on studies of operational experience. NRR is the headquarters

office for reactor licensing activities. A similar role for nonreactor licensed activities and facilities is conducted by the Office of Nuclear Material Safety and Safeguards (NMSS).

(c) Activities in the Office of Inspection and Enforcement (IE) include receiving prompt notifications (50.72 telephonic reports within one hour after the event) and providing prompt assessments of the safety significance of the event plus coordination and support of the immediate NRC response actions; making notifications of signficant events to other NRC offices; evaluating operating experience and conducting analyses of operational data to determine the need for IE actions (such as orders, bulletins, circulars, information notices, or enforcement activities); and coordinating with other NRC offices on activities associated with the overall operational data program.

(d) Activities in the Office of Nuclear Regulatory Research (RES) primarily are providing support, on an as needed basis, for immediate response to significant operational events and conducting longer-term research activities on generic issues and the implications of operating experience from a risk assessment and reliability standpoint.

(e) The regional offices' activities include providing an immediate response to significant operational events including onsite reviews to determine the need for additional NRC response to protect public health and safety; identifying issues or situations requiring IE headquarters or NRR actions (such as IE bulletins, license modifications, and generic evaluations); reviewing operational experience from a

compliance and Systematic Assessment of Licensee Performance (SALP) standpoint; and supporting the overall NRC operating experience program with inspection data and technical evaluations.

3.2 Results of Reviews of RTB Failures Conducted Prior to the Salem Events

The following are the results of reviews of RTB failure information prior to the Salem events:

(1) Reported LER data indicates that licensees over the years generally were identifying the cause of the RTB and UV device malfunctions and failures, implementing corrective actions (replacement, preventive maintenance, increased surveillance, etc.), and reporting to the NRC as required. Some licensees contacted the NSSS vendor or the manufacturer of the RTBs for assistance in resolving their problems. Some also reported RTB failures to NPRDS. Licensees also responded to the various NRC bulletins and circulars, NSSS notifications and manufacturers' notices as they deemed appropriate. The appropriateness of many of the above actions remains under review.

(2) The RTB manufacturers were aware of RTB failures that were called to their attention by licensees, the NRC, and NSSSs. However, there seems to be no continuous, broad program within the RTB vendors to collect and trend RTB failure data on the RTBs that they had supplied. Westinghouse and General Electric did, however, issue service letters as noted in Appendix E.

(3) Based on discussions with INPO, LERs and NPRDS data (including breaker failures) were being routinely monitored by INPO during 1981 and 1982. Their overall assessment was that plant specific problems were being addressed, that the breaker failure frequency was low, that the number of

failures in light of an increasing population gave no evidence of a disturbing trend, and that there was not a significant safety concern. As a result, no Significant Event Reports (SERs) or Significant Operating Experience Reports (SOERs) were deemed necessary prior to the Salem events.

(4) The results of NRC reviews of RTB failures prior to the Salem Unit 1 events are primarily reflected in Appendix E in terms of major actions taken on a generic basis. In addition, assessments by NRC offices of licensee operating experience reports dealing with RTB failures since 1980 are summarized below.

(a) AEOD - Based on the AEOD technical staff's screening of LERs, the St. Lucie Unit 1 event of November 30, 1980, was identified in 1981 as potentially significant and warranting further review. The St. Lucie Unit 1 event occurred during normal full power operation. The licensee discovered that several RTBs would not trip on undervoltage as expected, though the shunt trip devices were operable. The licensee thus identified a need to independently test the UV devices and shunt trip devices to assure their operability. Subsequently, AEOD conducted an engineering evaluation which resulted in a memorandum to IE recommending that a circular be issued to inform licensees of the event and to recommend that they revise their surveillance testing procedures for RTBs. The underlying concern was for reducing the potential for undetected failures of UV trip attachments. IE subsequently issued Circular No. 81-12, "Inadequate Periodic Test Procedure of PWR Protection System," on July 22, 1981, to all nuclear power facilities holding an operating license or a construction permit. The circular described the St. Lucie Unit 1 experience; discussed the need for independent testing of the UV device and the

shunt device; noted that the UV problems reported were the subject of IE Bulletin No. 79-09, dated April 17, 1979; and recommended revisions to testing procedures to include independent testing of each trip device.

The RTB failures that occurred at Salem Unit 2 in August 1982 and January 1983 were under review within AEOD at the time of the Salem Unit 1 ATWS events. However, the AEOD screening process did not see a trend or pattern from the review of individual LERs on RTB failures. Other reported RTB failures were categorized as single random failures that were within the licensing envelope, but pertinent to component failure rate studies. (Note: documentation is incomplete for 1980 and 1981.) Thus, an emerging trend or pattern was not recognized. The individual engineers saw few, apparently random and separate events and, coupled with the appropriate licensee corrective action, perceived a low failure rate for the RTBs and no generic safety concern. All RTB events in 1981 and later were being added to the Sequence Coding and Search System (SCSS) data base for subsequent analysis by a statistical trends and patterns analysis program that is being developed by AEOD.

Although the LERs for RTB failures were reviewed by the Lead Engineer for the affected plants, the central point in AEOD for reviewing RTB concerns (the Plant Systems Group) did not review all the LERs on UV devices and RTBs. However, even if they had reviewed all of the LERs, it is unlikely that actions would have been initiated beyond that taken on the November 1980 St. Lucie event. The principal reason

was that AEOD was aware that the ATWS Task Force was conducting an in-depth review of breaker failures, and it has been the AEOD policy not to perform in-depth studies of events or situations already receiving considerable attention by other NRC operating experience review groups. In this case, no need for an independent study by AEOD was apparent.

(b) NRR - Individual LERs such as those involving RTB failures were sent to about 15 branches for information. No generic problem was identified, and specific concerns, if any, were dispositioned on a plant-by-plant basis. Little documentation was available on the results of individual LER reviews by the various branches. However, it appears that RTB failures were generally viewed as single random failures with a perceived low overall failure rate. For example, the Salem Unit 2 event of August 20, 1982 was included in the September 8, 1982 operating reactor events briefing as an item of interest; however, no specific followup actions were developed.

The Salem Unit 1 trip on February 22, 1983 was under active review and was being discussed with IE personnel (see item c below) by the Operating Reactor Assessment Branch, Division of Licensing.

The Instrumentation and Control Systems Branch, Division of System Integration, issued a memorandum in November 1982 listing LER failure data for electrical and mechanical components within the RPS at operating reactors. Twenty-eight scram breaker failures occurring between 1973 and early 1982 were identified and reviewed as potential precursors to an ATWS event. A rough overall failure rate of 0.08 failures per reactor year was calculated, although the calculations

were not statistically rigorous and had a large uncertainty. The results were provided to RES, Reactor Risk Branch, for their use in conjunction with the ATWS rulemaking. No safety concern or significance was attributed to RTB failure data.

(c) IE Headquarters - IE issued IE Circular 81-12, as noted previously, in response to the St. Lucie event on November 30, 1980. The Events Analysis Branch (EAB) reviewed the LERs on RTB failures, but little documentation exists on the results of the reviews. EAB personnel, however, based on a review of the Salem Unit 2 LER 83-001/3L dated January 27, 1983, and the similar problem on August 20, 1982 which was referenced in the LER, initiated actions regarding the RTB failures. During the week of February 13, 1983, EAB contacted AEOD's Reactor Systems Group and suggested that the event be considered as a possible abnormal occurrence and that the cited failures be reviewed and analyzed for ATWS considerations. EAB also contacted Westinghouse personnel, as well as the preparer of the LER at Salem Unit 2 and the NRC resident inspectors at Salem Unit 2, to obtain additional information on the frequency for testing the breakers and to determine whether the shunt trip coils were used in any automatic trip modes. Unfortunately, the ATWS events at Salem Unit 1 on February 22 and February 25, 1983 occurred before the reviews were complete and corrective action taken.

On February 23, 1983, EAB personnel discussed the technical aspects of the February 22, 1983 event at Salem Unit 1 with NRR. The reported technical details were discussed and it was concluded that more information was needed to evaluate the event. Later EAB personnel independently pursued concerns regarding the apparent lack of an

automatic scram coupled with the recent RTB failures at Salem Unit 2. Subsequent erroneous information (i.e., obtained prior to the February 25 event) indicated that an automatic scram had occurred on February 22, and EAB's concern was thus alleviated. The review of the RTB malfunctions continued, however, and EAB indicates that it is likely that the breaker unreliability would have been discovered since the same personnel were involved as those who had prepared the earlier bulletins and the circular on RTB malfunctions. In addition, a potential generic issue concerning performance of the GE type AK-2 circuit breakers was sent to IE in October 1982 by the Region V office for information and was under review for potential generic action.

(d) RES - RES had not yet reviewed circuit breaker failures or failure rates using operational experience data. Circuit breakers, including the RTBs, were included among the components to be analyzed at some future date by both the LER Evaluation Program and the In-Plant Reliability Data System which contains records from GE, Westinghouse, and B&W plants. The Risk Assessment Branch was reviewing the information provided by NRR in November 1982 on LER failure data for electrical and mechanical components within the RPS, and was primarily interested in the ratio of electrical to mechanical failures in the RPS as it related to ATWS studies.

The ORNL precursor studies conducted under RES contracts using LER information also did not identify any RTB failures as precursors to core damage.

(e) Regional Offices - The regional offices performed onsite inspections and/or LER reviews for the reported RTB failures. The response actions

varied on a plant-by-plant basis. For example, the Salem Unit 2 RTB failures on August 20, 1982 and January 6, 1983 received considerable onsite review. The Senior Resident Inspector required LER 82-072 on the August 20 event to be supplemented because of incomplete descriptions of cause and corrective actions. The maintenance work subsequent to the January 6 failure was being followed. The fact that ongoing safety-related work was being accomplished without a procedure, though the Westinghouse representative was in the process of developing one for the licensee, was noted as a deficiency and placed on the inspector's "open items" list.

At several sites, the history of breaker failures was reviewed and NRC inspection reports note that the licensees indicated they were aware of ongoing breaker problems and were researching solutions including discussions with the breaker manufacturers.

Since 1981, the onsite inspection, followup, and close-out of licensee response actions on some IE bulletins, circulars, and information notices have received a low priority in NRC inspection activities. As a result, followup actions on Circular 81-12 were incomplete.

In October 1982, Region V had identified two problems with GE Type AK-2 circuit breakers as potentially generic and forwarded the inspection findings from Rancho Seco to IE headquarters and the other regional offices for their information and consideration. The information was under review at the time of the Salem Unit 1 ATWS events.

In summary, prior to the Salem Unit 1 ATWS events the NRC reviewed individual LERs involving RTB failures both in Headquarters offices and in the regional offices. The results of these assessments were that most individual RTB failure events were single, random failures, although common cause features such as UV device failures and inadequate maintenance practices, and a potential generic concern for the GE AK-2 circuit breakers were identified. The RTB failures at Salem Unit 2 in August 1982 and January 1983 were under review. However, a specific IE concern for a relationship between the RTB failures at Salem Unit 2 in January 1983 and the February 22, 1983 trip at Salem Unit 1 was temporarily alleviated based on the erroneous information that Unit 1 had tripped automatically.

3.3 Analysis of RTB Failure Data Conducted After the Salem ATWS Events

AEOD gathers information from LERs on a very wide range of operational experience. This approach results in the receipt of a large volume of routine reports (approximately 4,800 LERs were reviewed in 1982 and more than 13,000 since 1980), many of which, though screened for safety concerns, cannot be studied in-depth by AEOD, or other NRC offices, because of a lack of resources. As a result, the aggregated information is also to be reviewed for potential trends and patterns. An automated (computer-assisted) trends and patterns program is under development and is intended to be used to systematically search this massive volume of operational experience for increasing incidence or prevalence of particular events which would raise concern and perhaps, upon further investigation, lead to identification of potential safety problems and recommendations for corrective action.

3.3.1 Statistical Review of RTB Failures

After the Salem Unit 1 events, AEOD performed statistical analyses to demonstrate some of the techniques normally used in routine data analyses.

The statistical analyses performed were exploratory rather than rigorous. The objective was to see if the data would have led to conclusions or further study, had such analyses been done before the Salem Unit 1 events.

The approach used was to display the data in tabular and graphic form, make simple assumptions, group and transform the data, and analyze the results. In view of the limited data (lack of essential detail about how and when failures occur), no statistically rigorous parametric estimations or confirmatory analyses were performed.

To develop the data after the Salem Unit 1 events, LERs, Bulletin responses and the NPRDS were searched for instances where an RTB failed to trip (open) on demand (i.e., the failure modes of failure to reset (closed) and spurious opening were not considered). The basic data are compiled in Appendix B. Tables 3.1, 3.2, and 3.3, show the failure counts extracted from the data from 1971 through 1982 displayed in three two-way tables (i.e., by plant and calendar year for each of the three PWR NSSS vendors).

The RTBs, though very critical components of the important RTS, represent only a small portion of the safety-related components in a nuclear power plant. The RTB failure data available (about 50 failures) was only a small fraction (less than 0.2%) of over 30,000 LERs submitted to date.

Overview of Reported Failures

Examination of Tables 3.1 through 3.3 shows an imbalance in the reporting pattern. Only 20 plants, with a total accrued operating time (since being declared "commercial") of 156 operating years, have reported failures, while the total PWR population using RTBs with UV devices consists of 46 plants

Table 3.1

RTB Failure History by Plant
(Westinghouse NSSS)

	Plants	Number of Failures Reported*												
		71	72	73	74	75	76	77	78	79	80	81	82	
1)	Beaver Valley 1													
2)	Cook 1													
3)	Cook 2													
4)	Farley 1													
5)	Farley 2													
6)	Ginna													
7)	Haddam Neck**	2												
8)	Indian Point 2													
9)	Indian Point 3													
10)	Kewaunee						1		3					
11)	McGuire 1													
12)	North Anna 1											1	1	
13)	North Anna 2													
14)	Point Beach 1						1							
15)	Point Beach 2								2	1				
16)	Prairie Island 1													
17)	Prairie Island 2													
18)	Robinson 2	3		1								2	1	
19)	Salem 1													
20)	Salem 2												1	
21)	San Onofre 1													
22)	Sequoyah 1													
23)	Sequoyah 2													
24)	Surry 1													
25)	Surry 2											1		
26)	Trojan													
27)	Turkey Point 3													
28)	Turkey Point 4													
29)	Yankee Rowe 1***													
30)	Zion 1						1	1						Total=26
31)	Zion 2							1		2				

Year

* Reported Prior to 1/1/83.

** Uses both a UV trip and a shunt trip device. The UV devices failed during testing.

*** Does not use the UV device to trip the RTBs.

Table 3.2

RTB Failure History by Plant
(Combustion Engineering NSSS)*

	Plants	Number of Failures Reported**												
		71	72	73	74	75	76	77	78	79	80	81	82	
1)	Arkansas 2													
2)	Calvert Cliffs 1								1	1				
3)	Calvert Cliffs 2												1	
4)	Fort Calhoun 1***													
5)	Maine Yankee													
6)	Millstone 2													
7)	Palisades***													Total=6
8)	St. Lucie 1										3			

Year

* Both the UV device and shunt trip device are used.
** Reported prior to 1/1/83.
*** Do not use RTBs.

Table 3.3

RTB Failure History by Plant
(Babcock & Wilcox NSSS)

	Plants	Number of Failures Reported*												
		71	72	73	74	75	76	77	78	79	80	81	82	
1)	Arkansas 1								3				3	
2)	Crystal River 3								1					
3)	Davis Besse 1										1	1		
4)	Indian Point 1													
5)	Oconee 1		1							3				
6)	Oconee 2													
7)	Oconee 3					1				1	1			
8)	Rancho Seco												2	
9)	Three Mile Isl. 1					1	1		1					Total=21
10)	Three Mile Isl. 2													

Year

* Reported Prior to 1/1/83.

with a total accrued operating time of 334 operating years. Thus, 26 plants with a total of 178 accrued operating years show no failures. Two obvious explanations for the observed pattern are:

(1) specific, but unknown, reasons exist why there is not a more uniform distribution, and/or

(2) data are missing.

Failures by Vendor

From an analysis of the groupings by NSSS, a disparity was observed in reported failures between B&W plants on the one hand, and Westinghouse and CE plants on the other (see Table 3.4). This observation can be due to one or more of the following factors: (1) Westinghouse and CE plants are truly better performers, (2) some events at Westinghouse and CE plants are not being reported, (3) some events at Westinghouse and CE plants are not being recognized because of different testing procedures, and/or (4) there are differences in the design of the plants that contribute to the differences observed in the data (e.g., the number of breakers per plant is different). (In hypothesis testing jargon, the hypothesis that the rate of reported failure for B&W, Westinghouse and CE are equal can be rejected; this is true in fact at the 0.01 level of significance; and B&W plants contribute the most to rejection of the hypothesis.)

Table 3.4

Rate of Reported Failures/Reactor-Year by Vendor

Vendor	Failures	Reactor-Years	Reported Failures/Reactor-Year
B&W	21	61	0.34
CE	6	39	0.15
W	26	234	0.11

A more detailed review of the plant characteristics indicated that part of this disparity might be due to the use of reactor years as a basic measure of potential for having a failure reported. For the specific case of the RTBs, an intuitively better measure would result from multiplying the operating time for each reactor by the number of breakers at that plant to derive breaker operating years.

Using this basis to assess the reported failure pattern among the three NSSS vendors (see Table 3.5), yields the result that the difference among the vendors in rates is less statistically significant than in the previous cases (reject the homogeneity hypothesis at the 5% level, but not at the 1% level). An analyst might, however, have pursued the difference indicated by the point estimates. In fact, while the statistical analysis (based on a Poisson model) is not strong enough to reject the notion that CE is different for B&W and Westinghouse, review of testing practices leads to the strong suspicion that the CE numbers are not comparable because routine surveillance may not have independently tested operation of the UV trip device.

Table 3.5

Rate of Reported Failures/Breaker-Year by Vendor

Vendor	Failures	Breaker-Years	Reported Failures/Breaker-Year
B&W	21	356	0.059
CE	6	312	0.019
W	26	468	0.056

Failures by Plant

To get additional perspective on the pattern of known failure experience, rates of reported failures were calculated for each plant by summing the failures reported from 1971 through 1982 and dividing by the applicable number of breaker years. These rates would correspond to maximum likelihood estimates of the reactor trip circuit breaker "failure rate" if the following conditions are satisfied:

(1) Failure mechanisms are exponential, and the failure rate is constant over the period from 1971 through 1982.

(2) Failures are detected by surveillance demands which are evenly spaced.

For exploratory purposes, the observed failures for each plant are treated as the realizations of some Poisson process whose parameter may or may not correspond to the reactor trip circuit breaker "failure rate." The calculated rates are referred to as point estimates and are tabulated in Table 3.6.

Table 3.6

Rate of Failure for Plants Reporting

Westinghouse NSSS	Rate of Failure per Breaker Year
1) Haddam Neck	0.084
2) Kewaunee	0.240
3) North Anna 1	0.220
4) Point Beach 1	0.042
5) Point Beach 2	0.150
6) Robinson 2	0.300
7) Salem 2	0.400
8) Surry 2	0.053
9) Zion 1	0.110
10) Zion 2	0.190

Table 3.6 (continued)

Babcock & Wilcox NSSS	Rate of Failure per Breaker Year
1) Arkansas 1	0.130
2) Crystal River 3	0.029
3) Davis Besse 1	0.100
4) Oconee 1	0.068
5) Oconee 3	0.063
6) Rancho Seco	0.044
7) Three Mile Island 1	0.110
Combustion Engineering NSSS	
1) Calvert Cliffs 1	0.022
2) Calvert Cliffs 2	0.033
3) St. Lucie 1	0.063

The point estimates were then plotted in a version of a stem and leaf plot to get some perspective on how they group. These plots are shown in Appendix G, Figures G.1 through G.3. These plots are based on point estimates, not interval estimates which give some indication of uncertainty and tend to minimize any groupings. Therefore, care must be exercised in their interpretation. With this caveat in mind, the plots tend to show that within each NSSS, the NSSS average rate does not give a good summary (central tendency) for the individual plants. For example, the individual plant estimates for Westinghouse do not cluster around the Westinghouse average. A formal test for homogeneity within each group indicated that only for the Westinghouse group is there enough statistical evidence to reject the proposition that all the plants belong to a single population. This does not mean that B&W or CE plants do behave like members of a group, just that the evidence does not permit rejection of that hypothesis. It is still possible that the scatter observed in the point values is due to plant specific factors.

Rate of Reported Failure as a Function of Time

As a last step, the rates of reported failures were averaged across all plants for each year from 1971 through 1982 and this average rate was tabulated in Table 3.7 and plotted in Figure 3.1 with error bounds which bracket the average estimated rate. While this plot does not represent what any individual plant might do at any point in time, it gives a good indication of the behavior of the industry.

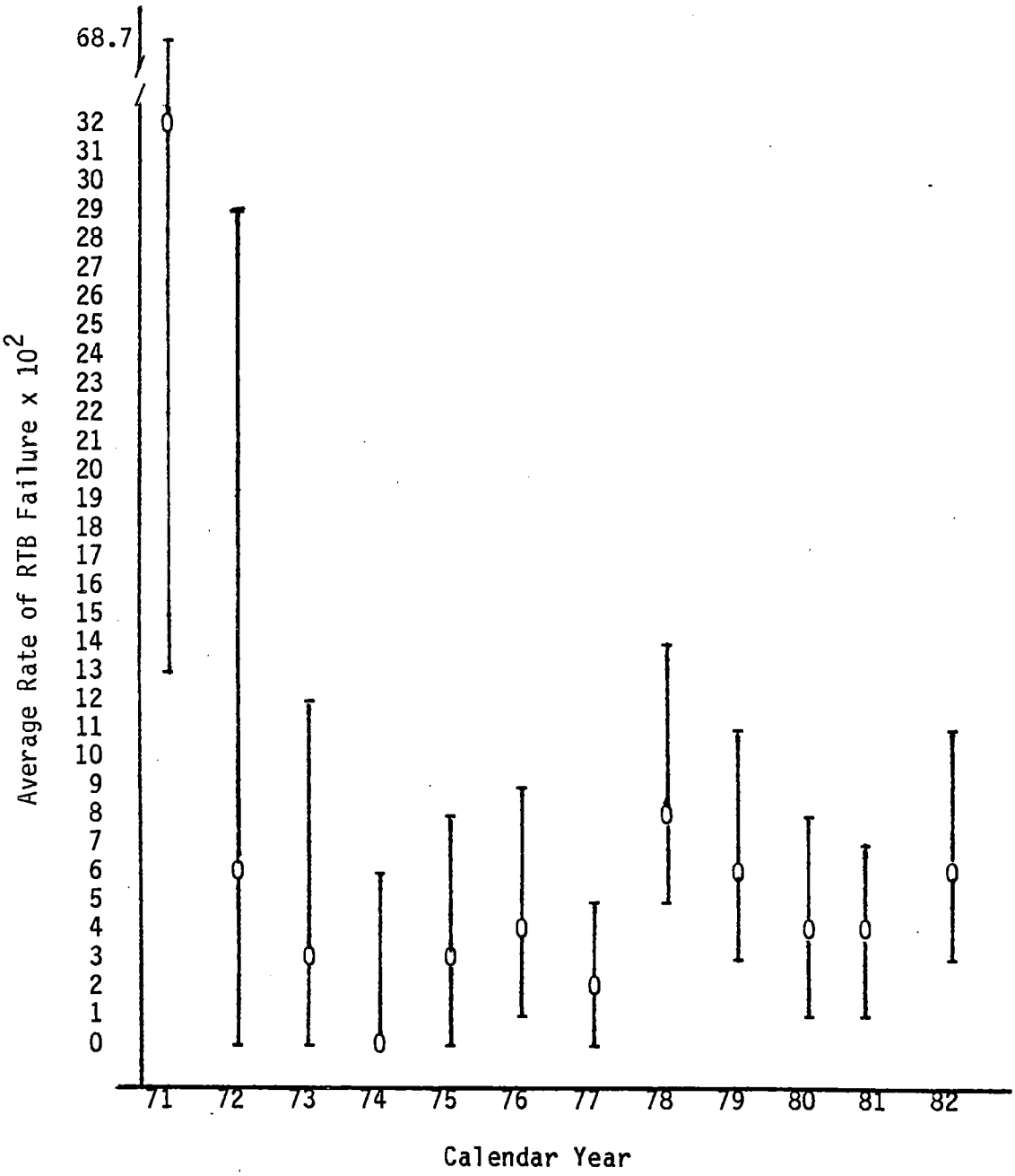
Table 3.7

Average Rate of Failure by Year

Year	Average Rate of Failure per Breaker Year
1971	0.32
1972	0.06
1973	0.03
1974	-
1975	0.03
1976	0.04
1977	0.02
1978	0.08
1979	0.06
1980	0.04
1981	0.04
1982	0.06

Figure 3.1

Average Rate of RTB Failure Versus Time



When the error bounds are taken into account, Figure 3.1 shows only one marginally significant "trend"; e.g., a shift upward in 1978 when looking at pairs of preceding/succeeding years, as an individual monitoring for short term indications would. A look at the entire time series shows that 1971 was significantly different from the years 1972-1982 taken as a group.

Reactor Trip Breaker Failure Rates

The foregoing was an exploratory analysis of failure reporting patterns. The data as collected supports the analysis directly. However, the matter which is of ultimate concern is determination of the probability of failure on demand of individual reactor trip circuit breakers. Estimation of this quantity requires additional considerations and assumptions.

A reactor trip circuit breaker is a periodically tested standby component. Surveillance testing is performed to determine if the circuit breaker has lost the capability to open, i.e., entered the failed state, at some point since it was last known to be operable. That surveillance testing is performed at all implies that time spent in standby plays some role in determining the circuit breaker's ability to perform, i.e., there is at least one significant time-dependent failure mechanism. Otherwise, all failures would be related to demand (test or actual) stresses and surveillance testing would be pointless. "Deterioration of coil's nylon sleeve," "dirt or corrosion on UV relay," "binding of linkage due to dirt or hard grease," and "dirty breaker" have been cited as failure mechanisms for reactor trip breakers and these strongly imply time dependence.

Thus, the distinction between failures caused by demand and failures discovered by demand is desirable. Clearly the basic data consists of failures discovered by demand, where the cause of the failure is probably a mixture of time dependent and cyclic (wear, stresses associated with a successful demand) mechanisms.

The quantity known is the number of failures historically discovered by demand, and the quantity desired is the probability of failure on operational (actual) demand. To get from the former to the latter requires some more detailed information, or lacking it, making some further assumptions.

The number of demands is needed, and, if there is a time dependence for the probability of being in the failed state, the spacing of the demands. Usually records of the number of demands are not kept, and this is true of the reactor trip circuit breaker case. In such cases, the usual procedure is to assume that the number of operational demands is negligible when compared to the number of test demands, and that the number of surveillance demands over the period for which failure data has been collected can be estimated using the maximum surveillance interval specified in standard Technical Specifications. Care must be used in "guesstimating" the number of demands since 1) the probabilities estimated may be sensitive to the number of demands used, and 2) the specific Technical Specifications surveillance requirement can be significantly lower than the actual number of tests.

The standard Technical Specifications call for bimonthly testing of reactor trip circuit breakers. However, not all plants have standard Technical

Specifications, and discussions during the Salem investigation indicated that some plants may test as often as monthly, but plant-specific numbers are not readily available. Thus, the number of surveillance demands is felt to be in the range of 6-12 per reactor operating year.

Figures are also not readily available on a plant specific basis for the number of actual demands made on reactor trip breakers. The average number of reactor trips per plant per reactor operating year is approximately eight. This number is not negligible when compared to the range of surveillance tests per year.

Table 3.8 shows reported failure counts and a range of demand counts grouped by NSSS as well as for all plants combined.

Table 3.8

Failure and Demand Counts by NSSS Vendor

Vendor	Failures	Breaker Years	Estimated Breaker Demands Test	Actual	Total
B&W	21	356	2,136-4,272	2,848	4,984-7,120
CE	6	312	1,872-3,744	2,496	4,368-6,240
W	26	468	2,808-5,616	3,744	6,552-9,360
All	53	1,056	6,816-13,632	9,088	15,904-22,720

Table 3.9 displays the ratio of reported failures to total estimated demands by NSSS Vendor and for all plants.

Table 3.9

Ratio of Failures to Total Demands by NSSS Vendor

Vendor	Failures/Demands
B&W	$3 \times 10^{-3} - 4 \times 10^{-3}$
CE	$1 \times 10^{-3} - 1 \times 10^{-3}$
W	$3 \times 10^{-3} - 4 \times 10^{-3}$
All	$2 \times 10^{-3} - 3 \times 10^{-3}$

The failure/demand ratio of Table 3.9 is a statistic which captures past performance under historical conditions. It does not necessarily represent the probability that a reactor trip breaker will fail to trip when presented with an actual demand. Consider the following:

(1) If there is no relationship between the time in standby and the probability of failing on demand i.e., no time dependent failure mechanisms of any consequence, then the ratio is a maximum likelihood estimate of the probability of failure in response to an operational demand.

(2) If there is a strong time dependence for failure probability, but both test and actual demands are made at random times, then the ratio is a time averaged estimate of the probability of failure in response to an operational demand.

(3) If surveillance testing is conducted at regular intervals and there are significant time dependent failure mechanisms, then surveillance failures and surveillance demands should be separated from failures discovered by actual demands. One then calculates an estimate of the probability of failure detected by a surveillance demand, <u>and</u> an estimate of the probability of a failure detected by an actual (random) demand.

Tables 3.10 and 3.11 display the results of proceeding as suggested in (3) above. However, while using only surveillance demands, one can argue that all failures count as surveillance failures, since failures occurring on actual demand would likely have been surveillance failures had the demand not occurred.

Table 3.10

Estimate of Probability of Failure on Test Demand

Vendor	Failures	Test Demands	Failures/Demands
B&W	21	2,136-4,272	$5 \times 10^{-3} - 1 \times 10^{-2}$
CE	6	1,872-3,744	$2 \times 10^{-3} - 3 \times 10^{-3}$
W	26	2,808-5,616	$5 \times 10^{-3} - 9 \times 10^{-3}$
All	53	6,816-13,632	$4 \times 10^{-3} - 8 \times 10^{-3}$

Table 3.11

Estimate of Probability of Failure on Actual Demand

Vendor	Failures	Actual Demands	Failures/Demands
B&W	3	2,848	1×10^{-3}
CE	3	2,496	1×10^{-3}
W	4	3,744	1×10^{-3}
All	10	9,088	1×10^{-3}

It is impossible to determine with the level of detail provided in LERs which set of failure probabilities, Table 3.9 or 3.11 is more representative of actual future performance. From the standpoint of comparison of reactor trip circuit breaker performance with published values, use of one set instead of the other does not alter the conclusion that performance is not outside the expected range:

(a) WASH-1400 gives a figure of 1×10^{-3} per demand (presumed random) for circuit breaker failure-to-transfer (i.e., failure to close or failure to open), with an error factor of 3 (i.e., a range of 3×10^{-4} to 3×10^{-3}.

(b) IEEE Standard 500-1977 gives a figure of 1.9×10^{-4} per demand with a range of 2×10^{-5} to 2×10^{-3}.

In making the comparisons between the figures in Tables 3.9 or 3.11 and WASH-1400 or IEE 500, two points must be kept in mind:

(a) The WASH-1400 or IEEE 500 numbers <u>are not standards</u> for performance of a reactor trip circuit breaker. They are other estimates of circuit breaker failure rates.

(b) The estimates shown in Tables 3.9 and 3.11 have error bounds of their own. Application of such error bounds to the B&W figure in Table 3.9, for example, would give a range which overlaps with the WASH-1400 range, and thus would blur any conclusion about B&W rates being "higher" than WASH-1400.

The figures in Table 3.10 are all high by comparison with the WASH-1400 nominal rate of 1×10^{-3}. However, the argument which lead to their calculation implied that such failure probabilities would only apply immediately prior to surveillance testing, and hence, they serve as a kind of upper bound over the test cycle with the probabilities of Table 3.11 representing the performance that would be seen by a random, intra-cycle demand. These numbers are very sensitive to the assumptions about demand history and, thus have a large uncertainty. This uncertainty can best be reduced through use of actual test frequencies which are not available through LER or other routine reports submitted to the NRC.

3.3.2 Trends and Patterns Review

From 1971 to the present, the record of NRC and industry actions on RTBs indicates that an ad hoc trends and patterns program existed and was prompting corrective action (see Appendix E). The failure record discussed in detail previously shows that at an individual plant level there are no definite trends -- only isolated, grouped failures. Thus, the only reasonable basis for action based on the data at the plant level would be investigation of the clustered failures.

Major NRC action was taken in 1979 in response to the detection of a trend across both plants and time. IE Bulletin No. 79-09 indicated nine failures of General Electric (GE) type AK-2 (i.e., AI-2A-15, 25, 50, 75, 100) circuit breakers installed in safety-related systems had been reported since 1975. The causes of failure were attributed to either binding within the linkage mechanism of the undervoltage (UV) trip device and trip shaft assembly or out-of-adjustment conditions in the same linkage mechanism. Babcock and Wilcox (B&W) and GE determined that the binding and out-of-adjustment conditions resulted from inadequate preventive maintenance programs.

Analysis of the calculated failure trend on Figure 3.1 also showed that a marginally significant shift upward occurred in 1978. The data include, however, a number of failures which were not included in the Bulletin study.

As mentioned previously, a computer-assisted trends and patterns program is being developed by AEOD, which will focus on data at a fairly high level of aggregation. For example, one analysis might deal with event counts categorized according to the following variables:

COMPONENT: Circuit Breaker (ac or dc)

PLANT: All plants with operating licenses

TIME: 12 divisions corresponding to each calendar month in 1981

Because of the unavailability of component population counts, the unavailability of accurate data regarding the number of demands and time in service, and the difficulty of automating that information in the analysis even if known, AEOD would normalize the count data by reactor time and not component time or demands. Accordingly, Table 3.12 displays the unnormalized counts of ac or dc circuit breaker faults reported in an LER. An asterisk next to a number indicates that the number includes one of the three 1981 occurrences (as reported by LERs) of an RTB failure to open. Since only three of 206 total occurrences reported by LERs involved RTBs, it is doubtful that such analysis would lead an analyst to focus on the reactor trip area. The best that could be expected would be to duplicate the process which led to issuance of Bulletin No. 79-09; that is, in looking across time and plant, the analyst might identify a trend, pattern or anomaly which would lead to investigating a type of circuit breaker which also happens to be used in the reactor trip function.

The planned program would include generating tables similar to Figure 3.10 for major components and subsequent analysis of the information.

3.3.3 Inference of Common-Cause Failure Probabilities

Routine review of individual reports provided qualitative indication of common-cause failure potential. Also, the 1971 event at Haddam Neck and the 1980 event at St. Lucie Unit 1 were instances of multiple failures possibly due

Table 3.12

Counts of ac or dc Circuit Breaker Faults*
(Faults/Month in 1981)

Plant	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Plant Total
Arkansas 1								9					9
Arkansas 2											2		2
Davis-Besse 1	2			1		2	1			4**	1		11
Cook 1			5				1		1			1	8
Cook 2	1				1							1	3
Ft. Calhoun 1				1									1
Calvert Cliffs 1							1						1
Calvert Cliffs 2	1	1	1	2		1	1						7
Crystal River 3	4		1										5
Beaver Valley 1						1		1	1	1		9	13
Haddam Neck 1				1									1
Indian Point 2						2		1					3
Farley 1	1	9		1				2					13
Kewaunee 1	1		1	2				1		1			6
Millstone 2												9	9
Maine Yankee 1						1							1
North Anna 1	1	1		2			1	2**					7
North Anna 2						1	2					1	4
Oconee 1	4					1	2						7
Oconee 3							1						1
Palisades 1	2						1						3
Pt. Beach 1	1			1				2					4
Prairie Isl. 1							3			1		1	5
Quad Cities 1	2		1										3
Quad Cities 2					3								3
Ginna 1	1		2	1							1		5
St. Lucie 1	1								1		3		5
Rancho Seco 1		12								1		1	14
Salem 1					3						1		4
Salem 2				2		1	6				1		10
Surry 1								1					1
Surry 2	1**				2		3	1					7
Three Mile Is. 1	1												1
Trojan 1	1			1									2
Turkey Pt. 3					1								1
Turkey Pt. 4			18				1						19
Zion 1	1		1	2	2								6
Zion 2									1				1
Monthly Total	26	23	30	17	12	10	24	21	3	8	9	23	206

* Any circuit breaker, not just RTBs.

** Includes an RTB failure.

to common cause mechanisms. However, no attempt was made to quantify the multiple failure probability, a quantity which, if large, might have added weight to an argument for corrective action and in some sense "predicted" the Salem Unit 1 events. EG&G Idaho, has developed for the NRC a computer code suited to this task. However, with the limited time available for this study, results from the code were not available to see if it provides useful output for the RTB case.

3.4 Conclusions

3.4.1 NRC's Operational Data Assessment Program

(1) In 1971, 1979, and 1981, the review of LERs by the NRC identified specific concerns regarding the RTB and related UV trip device failures. As a result, these concerns were addressed by regulatory actions (the issuance of IE Bulletin 71-2 on Westinghouse DB series RTBs, IE Bulletin 79-09 on General Electric AK series RTBs, and IE Circular 81-12 on surveillance testing of RTBs). In addition, the equipment vendors also issued notifications regarding the proper care of RTBs (Westinghouse Technical Bulletin NSP-TB-74-1 and NSD Data Letter 74-2; General Electric Service Advice Letter 175 (CPDD) 9.3). Thus, specific safety problems had been identified with these components through the review of operational experience. However, the common cause failure potential and broader generic implications of the individual RTB failures were not fully appreciated.

(2) A review of the actions taken and in progress within the various offices of the NRC indicates that the NRC's operational assessment program was being implemented and a number of RTB problems had been recognized and studied. For example: (a) the need to test the UV trip attachment separately from the shunt attachment was identified by AEOD and formally issued as regulatory guidance by IE; (b) generic concerns with the GE type AK-2-25 breakers were recognized by Region V and formally documented; (c) specific problems with the preventive maintenance program for the RTB breakers at Salem were recognized by the resident inspector and formally documented in an inspection report; (d) the August 1982 RTB failure at Salem Unit 2 was recognized a potential safety concern within NRR and discussed with NRR management and other NRC personnel during a periodic briefing; (e) the potential importance of the January 1983 RTB failure at Salem Unit 2 had been recognized by IE and to a lesser degree by AEOD and NRR, and was being actively investigated; and (f) a comprehensive study was made of RTB failures and their contribution to an ATWS event was assessed in 1982 by NRR in support of the ATWS Task Force activities. Thus, the various offices were involved and informed, but an overall program to address the safety concern was still in development (i.e., ATWS rulemaking).

(3) The Salem event again clearly emphasizes that operational data assessment requires clear and in-depth licensee reports; the ability to recall past failure histories with precision and completeness; the ability to analyze the failure history with regard to such aspects as the frequency, age, and location (specific plants involved); and the ability to identify the application of safety-related equipment which may be subject to the same

types of failures. As a result, there is even a greater incentive for the type of reports required by the proposed LER rule and by the NPRDS; improved and fully effective LER and NPRDS data bases that can be accessed and searched by all organizations involved with operational data assessment; a comprehensive and systematic trend and pattern analysis program; and a current and complete NPRDS engineering data file.

3.4.2 Trends and Patterns

(1) Because of (a) the observed failure rate of RTBs, which was about as expected, (b) the apparent random nature and low frequency of RTB failures at specific plants, (c) the multiple modes of failure and operating difficulties experienced with these breakers, and (d) the great volume of failure data of safety-related breakers and RPS components, it is highly doubtful that a statistically based trend and pattern analysis program alone would have identified the RTB failures as "outliers" or a safety problem worthy of specific in-depth engineering investigation. Such failures are indeed a problem but in view of the characteristics of this situation, the identification of this particular problem by the NRC would more likely be (and was to a degree in this case) a result of reviewing 10 CFR 50.72 and daily regional and LER reports from an engineering perspective. Trending of the breaker operational performance parameters such as the time for breaker operation or tripping force, by licensees should, however, identify incipient failures.

(2) Trends and patterns analyses alone can only provide guidance on where to place engineering resources to further investigate "outliers" and irregularities in operating experience data. However, planned trends and

patterns analyses will improve the routine display of collections of information and when coupled with close scrutiny of failure data and detailed engineering assessment, particularly of those features related to reliability, can identify specific plant and/or generic safety problems and the need for corrective actions.

(3) Over the years, safety concerns for concurrent RTB failures in a redundant RTB system using only UV trip devices were tempered by the lack of a clear or persistent increasing trend of RTB failures and the fact that the overall failure rate was consistent with expected performance for circuit breakers. Actions taken on a plant-by-plant basis, and several times on a generic basis, were apparently sufficient to maintain an acceptable failure rate. However, followup actions were sometimes not as long-standing or as effective as planned. For example, at least one plant did not continue to test the UV device separately as requested in IE Circular 81-12; and previous guidance and lessons of experience, such as identified in IE Bulletin 71-2, seem to have been lost and forgotten with time.

(4) Based on calculations performed after the Salem ATWS events, the individual plant data do not cluster well around the NSSS averages. However, except in the case of Westinghouse, the statistical evidence is not strong enough to say that plants do not behave like members of an NSSS-defined population. In reality, the problems may be a function of conditions at individual plants and, consequently, emphasis must be given to a close review of the failure rates and modes at individual plants. The calculated estimates of rate of RTB failure based on reported RTB failures were within the uncertainty bounds of previous estimates, i.e., were essentially as predicted and thus acceptable.

(5) Even though the events at Salem involved no plant damage, no releases, and no immediate threat to public health and safety, the fact that the NRC and the industry have devoted extensive resources to studying its cause and implications is a strong indication of the heightened sensitivity to operational events and the progress made in understanding the lessons of operational experience. The Salem events could be considered precursors of very serious events that, it is hoped, will never happen because of the attention paid to the Salem events and the corrective actions that are being taken.

4.0 Requirements for Data Retention, Event Analysis, and Operational Experience Feedback

4.1 Discussion

Operational experience is one of the most fundamental and best indicators of potential safety problems. To best utilize this experience, an in-depth program must be established and implemented which gathers, analyzes, and disseminates information on operational events. The events at Salem Unit 1, especially that of February 22, 1983, emphasize this need.

As noted earlier in this report, nuclear power plant licensees have requirements to assess operating experience, prepare the reports of off-normal events, take actions as necessary to correct unsafe conditions, and communicate specific types of experiences to the regulatory authorities. These requirements are contained in both NRC regulations, license conditions and technical specifications. Regulatory guides and industry standards have been developed to help clarify and add uniformity to the requirements.

However, as discussed in subsequent sections, the guidance given is fundamentally general in nature. As a result, specific methods to fulfill the requirements are left to each licensee and, consequently, practices differ widely from plant to plant.

4.1.1. Requirements for Event Analysis and Operational Experience Feedback

The fundamental document establishing the need for data retention, event analysis and operating experience feedback is 10 CFR 50, "Licensing of Production and Utilization Facilities." Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 establishes quality assurance requirements for the operation of nuclear power plant safety-related structures, systems, and components. Two sections apply directly to operational events: Criterion XVI, "Corrective Action" and Criterion XVII, "Quality Assurance Records."

In Criterion XVI it is stated that "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." In the case of significant conditions which are adverse to quality, Criterion XVI notes that "the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Criterion XVII supports this by stating that "Sufficient records shall be maintained to furnish evidence of activities affecting quality." Thus, these criteria give the basic requirements that any condition (be it component fault, personnel error, reactor trip, major loss-of-coolant accident, etc.) that may affect the quality of a nuclear plant must be studied, analyzed and, if necessary, corrected, and that records shall be kept of such activities.

To help guide a licensee on how to implement these requirements, several NRC and industry documents have been produced. The basic NRC documents covering operating reactors are Safety Guide 33, "Quality Assurance Program Requirements (Operation)," and subsequent revisions entitled Regulatory Guide 1.33. These guides endorse corresponding versions of the American National Standards Institute/ American Nuclear Society standard ANSI N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," dating from 1972 to 1982. These documents address the necessity that procedures be developed and followed for: (1) delineation of responsibilities for operational experience review; (2) routine and other plant operations including post-trip recovery and emergency and maintenance activities; and (3) audit and independent review functions.

For reactor trips, the standards state that the responsibilities and authorities of the plant operating personnel shall be delineated including the responsibility to determine the circumstances, analyze the cause, and determine that operations can proceed safely before the reactor is returned to power after a trip or an unscheduled or unexplained power reduction.

With regard to records management, they note that provisions should be made for preparation and retention of appropriate plant records and that the retention periods should be of sufficient length to assure the ability to reconstruct significant events and meet statutory requirements.

The NRC Standard Technical Specifications issued for the four light-water reactor vendor plants (NUREG-0103, -0123, -0212, -0452) call for the establishment of independent review committees to assess operational experience. Included in these committees' tasks is the review of significant operating abnormalities or deviations from normal or expected performance of plant equipment that affects

nuclear safety. They are also called upon to review any indication of an unanticipated deficiency in some aspect of the design or operation of safety-related structures, systems, or components.

After the Three Mile Island Unit 2 accident in 1979, further emphasis was placed on operational event assessment and feedback of operational experience. In NUREG-0737, "Clarification of TMI Action Plan Requirements," item I.C.5 tasks the operating plant licensee to prepare procedures to assure that operating information pertinent to plant safety be continually supplied to the operators and other personnel. Part of this feedback involves the assessment of operating experience, including reactor trips, which originates from many sources including the licensee's facility and outside sources.*

Concerning information from outside sources, the NRC has endorsed utility use of the Institute of Nuclear Power Operations (INPO) Significant Event Evaluations and Information Network (SEE-IN) program. This program provides a mechanism for centralized industry collection and screening of events from both U.S. and foreign nuclear plants. Participation in SEE-IN, however, does not relieve a utility from taking actions specific to the utility's nuclear unit which result from an evaluation of operational experiences. Each utility is still required to have an internal procedure for handling operational experience information, including the procedures necessary to assure that appropriate individuals are provided the results of evaluations and that recommendations for corrective action identified as a result of evaluation are translated into actions and that a record of such actions is retained.

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* Appendix H contains the specific requirements of I.C.5.

In all of the above documents, however, little guidance is given as to what constitutes a good program; how the analysis of an event, such as a reactor trip, is to be conducted prior to restart; what level of detail should be required of the analysis; what factors must be considered; and most importantly who has final responsibility to determine that the plant is ready to resume or continue operation. Because of the lack of specificity, and because of varying dates for implementation of the guides and standards in plant licenses and technical specifications, licensee programs are not uniform or very detailed in most cases.

As an example of how a utility attempted to meet the regulatory requirements, NUREG-0977, "NRC Fact-Finding Task Force Report on the ATWS Events at Salem Nuclear Generating Station, Unit 1, on February 22 and 23, 1983," discusses the methods employed at Salem for event review and assessment. Highlighted in NUREG-0977 are Salem administrative procedures for post-trip reviews and how these procedures were followed after the February 22 trip event. The licensees' program, though it conformed to the general nature of the guides and standards, lacked sufficient specificity to assure that the available plant data was thoroughly analyzed; i.e., the ATWS event on February 22 was overlooked in the post trip reviews, in part due to operations personnel being unfamiliar with how the computer processes the data. Printouts from the sequence of events recorder for reactor trips that occurred earlier in January and February 1983 were also disposed of after a short retention period. The lack of data precluded a complete reconstruction of pertinent operating history.

4.1.2 Data Requirements for Post-Event Analysis

Complete, detailed and accurate event analysis, whether during or after occurrences, is totally dependent on the quantity and quality (sufficient and

valid) of the data. Basic criteria documents which establish data collection and retention requirements are (1) Criterion 13, "Instrumentation and Control," of Appendix A to 10 CFR Part 50, which states that instrumentation shall be provided to monitor variables and systems over their anticipated ranges including normal operations, anticipated operational occurrences, and accident conditions; and (2) Criterion XVII, "Quality Assurance Records," of Appendix B to 10 CFR Part 50, which states that sufficient records shall be maintained to furnish evidence of activities affecting quality.

To meet these requirements, extensive guidance (Regulatory Guides 1.89 and 1.97, NUREGs-0588 and 0696, and ANSI/ANS-4.5-1980) has been developed for the equipment and data needed to monitor and assist the operators during the course of an event while it is occurring. Similar guidance for post-event analysis of occurrences, such as reactor trips, is not specifically defined.

To date, the understanding and reconstruction of operating experience has used available data. Where valid, detailed data were available, a precise and accurate description of the nature, course and cause of the event could be determined. However, when such information was not available, the ability to understand the event and to recognize important details of operational anomalies was hampered. The lessons to be learned and the feedback to prevent recurrence suffered due to uncertainty and misunderstandings.

An example of an event where adequate information, even if it was not complete, was available for event analysis is the Salem Unit 1 event on February 22, 1983; i.e., the sequence-of-event recording (taken from the plant process computer dedicated output typewriter) provided the only record that an ATWS event occurred. Examples of events where event analysis suffered because of the lack of sufficiently detailed and complete information are:

(1) Both units at Arkansas Nuclear One experienced a total loss of offsite power event on April 7, 1980. Both units established natural circulation, and plant response was basically as expected. However, post-event analysis was hampered due to the fact that the process computers for both units were unavailable during the transient. One was out because it was aligned to receive only ac power; the other one was out for unknown reasons, but it is thought that it tripped off due to a voltage decrease. Unit 1 had another computer for recording trends of variables and for performing calculations. The information from this computer would have been useful, but it also was not in operation. Consequently, a precise determination of alarms, equipment operations, or sequence-of-events was not possible.

(2) On February 26, 1980, Crystal River Unit 3 experienced a plant transient which was initiated by a loss of instrument power. During the event, power to the Non-Nuclear Instrumentation and Integrated Control System was lost, a small loss-of-coolant was created, and almost all of the control room instrumentation was lost for a period of about 20 minutes. During the course of this event, so much data were being generated that the sequence recording computer input buffer was overloaded, resulting in the loss of pertinent data. Thus, event reconstruction was hindered.

(3) During the January 25, 1982 steam generator tube rupture event at R. E. Ginna, the process computer was functional. This computer, however, was designed to read and record information important to the normal steady-state operation of the plant. It was not well-suited for post-event evaluation requirements since it did not record vital parameters such as safety-injection flow, safety system tank levels, and safety system valve positions. These limitations hindered some aspects of the post-event analysis.

(4) On July 26, 1980 following a scram of Hatch Unit 2, plant personnel found that the two of the four scram discharge volume high level switches were inoperable. An investigation revealed that both switches failed due to crushed floats. While attempting to determine precisely when damage to the floats had occurred, a review of process computer alarm edits revealed that the A and D float switches had not tripped during the three most recent scrams on June 2 and 14, 1980 and July 11, 1980. Further review of the process computer output of even earlier reactor trips was able to show that the switches had tripped and reset properly as late as during a scram on May 15, 1980. This was the last scram for which the switches could be verified operable. Although a subsequent scram had occurred on May 21, 1980, the process computer was not functional for approximately 21 minutes immediately following the scram. As a result, alarm edit records, which indicate the performance of the level switches, were not available for evaluation. The licensee concluded that the failures most probably occurred during the reactor scram on May 21, 1980, when the process computer was not available to record the functioning of the switches. Thus, in this case, the process computer data both helped and hindered the event analysis. The importance of adequate data retention is noted, however.

As indicated by the above examples, existing operational experience data acquisition for post-event analysis may not be adequate. Criteria seems to be needed to assure that each licensee gathers, records and analyzes data in sufficient detail, completeness and accuracy so that a thorough, valid technical understanding of each operational event, including any unusual behavior, can

be obtained first by the licensee and second, if necessary or desired, by other organizations such as the NRC or INPO. Criteria would be needed for items such as power requirements, number and scan-rates of parameters monitored, availability factors, and plant modes when the recording equipment must be operable to assure the availability of quality post-event reconstruction data.

4.2 Conclusions

(1) Operational event analysis and feedback by each licensee, the industry, and the NRC are essential for the safe operation of nuclear power plants. Several rules, regulations, and guidelines address this topic; however, they are not specific and, as a result, licensee activities vary widely. Although there are requirements that each licensee have procedures to carry out analysis and evaluation activities, each licensee makes an individual judgment regarding such items as what the elements of the program are; who is to be responsible for the program; to what degree independent evaluations are needed; what operational data is needed for the analysis program; what reviews are to be conducted and when; to what extent each event needs to be reconstructed; what constitutes a sufficient "understanding" of an event to allow resumption of plant operations; what weight should be given to long-term implications versus short-term plant operations consideration; how the results of any analyses are to be fed back to operations personnel and incorporated into operational documentation; and how the involvement of management is increased with the seriousness or implication of events. Licensee programs to evaluate and analyze operational events, particularly reactor trips, need to be systematic and thorough to ensure available data are properly analyzed, significant

failures are identified, the safety implications fully assessed, and proper corrective action completed.

(2) In order to perform an adequate post-event analysis and to properly identify anomalous behavior, events must be accurately reconstructed, including identifying initiating failures, causes for equipment operation or maloperation, activation of equipment alarms, operator actions involving equipment, and changes in plant parameters. A sequence of event recorder or combination of recorders is essential to provide the detailed information necessary to fully understand the cause, implications, and seriousness of an event. Personnel must also be trained to properly interpret and use the data and operating procedures established to ensure proper implementation.

(3) The NRC and the industry have spent considerable time and effort defining the information needed by the operators to follow the course of a serious event during the event. However, the requirements for collection and storage of data to support a posteriori routine event reconstruction are not well defined and have been, at best, a peripheral issue in the assessment of the needs of the operators during the event. It is clear from past events (the Three Mile Island accident, the Salem Unit 1 ATWS events, and the Arkansas Nuclear One loss of offsite power event) that a systematic assessment of the unique needs of post-event reconstruction is needed. Such aspects as what information is to be recorded; scanning and recording rates; quantity of data recorded and retention periods; and the requirements for equipment availability, reliability, and qualification; need to be specifically addressed either by other organizations (such as INPO or ANSI) or by the NRC.

Appendix A

Typical Technical Specification Section
on Reporting Requirements

1. Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the cognizant NRC Regional Office unless otherwise noted.

1.1 Routine Reports

1.2 Reportable Occurrences

Reportable Occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of an occurrence. In case of corrected or supplemental reports, reference shall be made to the original report date. [These reporting requirements apply only to Appendix A (of the license) Technical Specifications.]

a. Prompt Notification With Written Follow-Up. The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, telecopy or facsimile transmission to the Administrator of the cognizant NRC Regional Office, or his designate no later than the the first working day following the event, with a written follow-up report within two weeks. The written follow-up report shall include material to provide complete explanation, cause of the event, the circumstances surrounding the event, any corrective action, and component failure data.

1. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the Technical Specifications or failure to complete the required protective function.

   Note*: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under items 1.2.a.5, 1.2.a.6, or 1.2.b.1 below.

2. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications.

   Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the Technical Specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under item 1.2.b.2 below.

3. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

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* The "Notes" shown in this Appendix are not included in all licensees' technical specifications.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in the Technical Specifications need not be reported under this item.

4. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the Technical Specifications; short term reactivity increases that correspond to a reactor period of less than 5 seconds or, if sub-critical an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

5. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the FSAR.

6. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the FSAR.

Note: For items 1.2.a.5 and 1.2.a.6 reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under items 1.2.b.2 and 1.2.b.3.

7. Conditions arising from natural or man-made events that, as a direct result of the event required plant shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

8. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the FSAR or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the safety analyses.

9. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the FSAR or Technical Specifications bases; or discovery during plant life of conditions not specifically considered in the FSAR or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

   Note: This item is intended to provide for reporting of potentially generic problems.

b. Thirty-Day Written Reports. The reportable occurrences discussed below shall be the subject of written reports to the Administrator of the cognizant NRC Regional Office within thirty days of occurrence of the event. The written report shall include narrative

material to provide a complete explanation of the cause of the event, circumstances surrounding the event, any corrective action, and component failure data.

1. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

2. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

   Note: Routine surveillance testing, instrument calibration, or preventive maintenance which require system configurations as described in items 1.2.b.1 and 1.2.b.2 need not be reported except where test results themselves reveal a degraded mode as described above.

3. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

4. Abnormal degradation of systems other than those specified in item 1.2.a.3 designed to contain radioactive material resulting from the fission process.

   Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within

the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

1.3 Unique Reporting Requirements

(Plant Specific)

Appendix B
Licensee Reports on Reactor Trip Breaker Failures
(Prior to February, 1983 Salem Unit 1 Events)

Plant	Docket 50-	LER No.	2 week (T) or 30 day (L)	Date of Event	During Surveillance (S) or Operational Event (O)	Basis for Reporting Appendix A	Comments (1)	Cause of Failure	Breaker Vendor
Arkansas 1	313	78-23	T	9/25/78	S	1.2.a		Trip mechanism out of adjustment.	GE
Arkansas 1	313	78-24	L	9/16/78	O	1.2.a		Adjustment.	GE
Arkansas 1	313	78-29	L	10/17/78	S	1.2.b	Breaker tripped 1 minute late.	Unknown.	GE
Arkansas 1	313	82-16	L	7/14/82	S	1.2.b.2		Unknown.	GE
Arkansas 1	313	82-22	L	8/7/82	S	1.2.b.2		UV device out of adjustment.	GE
Arkansas 1	313	82-24	L	11/4/82	S	1.2.b.2		Latch improperly adjusted.	GE
Calvert Cliffs 1	317	None+	NPRDS submittal	2/28/78	*	+		Out of adjustment.	GE
Calvert Cliffs 1	317	None+	NPRDS submittal	5/3/79	S	+		Defective UV device.	GE
Calvert Cliffs 2	318	None+	NPRDS submittal	1/14/82	S	+		Defective UV relay.	GE
Crystal River 3	302	78-9	L	1/22/78	S	1.2.b.1		Undercurrent relay.	GE
Davis-Besse 1	346	80-80	L	11/3/80	S	1.		Unknown.	GE
Davis-Besse 1	346	81-70++	L	10/26/81	S	1.	Breaker may have (2) failed more than once.	Component failure of trip breaker.	GE
Haddam Neck (Connecticut Yankee)	213	71-7	T	12/2/71	S	*	Two failures.	Dirt on exposed linkages and rough surfaces.	W

Plant	Docket 50-	LER No.	2 week (T) or 30 day (L)	Date of Event	During Surveillance (S) or Operational Event (O)	Basis for Reporting Appendix A	Comments (1)	Cause of Failure	Breaker Vendor
Kewaunee	305	None+	NPRDS submittal	3/25/76	*	+		Binding on UV coil linkage.	W
Kewaunee	305	78-34	L	10/24/78	S	1.	Failed once on 10/24/78 & once on 10/31/78.	Normal wear of reset lever.	W
Kewaunee	305	78-37	L	12/26/78	S	1.		Sticking of shaft.	W
North Anna 1	338	81-63	L	8/8/81	S	1.2.b.2		Failed latch in UV trip attachment.	W
North Anna 1	338	82-73	L	11/18/82	S	1.2.b.2		Sticking of UV tripping mechanism.	W
Oconee 1	269	*	T	8/13/72	O	*		Spring tension and coil resistance too low.	GE
Oconee 1	269	79-5	L	1/22/79	S	1.2.b.2	Breaker failed once on 1/22, 3 times on 1/31 and once on 2/8.(3)	Trip mechanism out of adjustment.	GE
Oconee 3	287	75-11	T	8/7/75	S	*	Breaker failed several times on 8/7/75.(2)	Unknown.	GE
Oconee 3	287	79-2	L	1/18/79	S	1.2.b.2		Trip mechanism out of adjustment.	GE
Oconee 3	287	80-19	L	11/26/80	S	1.2.b.2		Sticking of trip shaft bearing.	GE

Plant	Docket 50-	LER No.	2 week (T) or 30 day (L)	Date of Event	During Surveillance (S) or Operational Event (O)	Basis for Reporting Appendix A	(1) Comments	Cause of Failure	Breaker Vendor
Point Beach 1	266	76-11++	T	11/30/76	S	*		Sticky UV device.	W
Point Beach 2	301	78-8	L	7/5/78	S	1.2.b.1	Breaker failed once on 7/5 and once on 7/17.	Deterioration of coil's nylon sleeve.	W
Point Beach 2	301	79-2	L	2/27/79	S	1.2.b.1		Deterioration of coil's nylon sleeve.	W
Rancho Seco	312	82-19	L	7/28/82	S	1.2.b.2	Breaker failed once on 7/28 and once on 7/30.	Linkage arm out of adjustment.	GE
Robinson 2	261	*	T	5/14/71	S	*	Three failures, each involving different problems with UV trip attachment, occurring on 5/14, 8/19, and 9/24/71.	Latch pin broke on one, binding on second, not specified on third.	W
Robinson 2	261	73-72	T	12/21/73	S	1.2.a	Breaker failed once on 12/21 and twice on 12/26.(4)	Excessive friction.	W
Robinson 2	261	None	Info. Report	9/23/81	S	Not given	Breaker failed twice.	Binding due to normal wear and dirt.	W
Robinson 2	261	None	Info. Report	12/20/82	S	Not given		Sticking UV relay.	W
Salem 2	311	82-72	L	8/20/82	S	1.2.b.2		Binding of UV coil solenoid.	W
Salem 2	311	83-1	L	1/6/83	O	1.2.b.2		Dirt or corrosion on UV relay caused binding.	W

Plant	Docket 50-	LER No.	2 week (T) or 30 day (L)	Date of Event	During Surveillance (S) or Operational Event (O)	Basis for Reporting Appendix A	(1) Comments	Cause of Failure	Breaker Vendor
St. Lucie	335	80-67	T	11/30/80	O	1.2.a.9	One breaker did not trip and two tripped sluggishly.	Out of alignment for one, sticking of other two.	GE
Surry 2	281	81-4	L	1/7/81	S	1.2.b.2		Missing bushing in UV device.	W
Three Mile Isl. 1	289	75-13	T	4/25/75	O	*		Mechanical linkage out of adjustment.	GE
Three Mile Isl. 1	289	76-30	L	9/1/76	S	1.2.b.2		Binding of linkage due to dirt or hard grease.	GE
Three Mile Isl. 1	289	78-9	L	3/15/78	S	1.2.b.2		Binding of linkage.	GE
Zion 1	295	76-53++	L	9/17/76	S	1.2	Breaker failed twice.(2)	Dirty breaker.	W
Zion 1	295	77-34	L	5/31/77	O	1.2	Breaker failed twice.(5)	UV trip relay not exerting enough force.	W
Zion 2	304	77-10	L	3/27/77	O	1.2.b.2		Unknown.	W
Zion 2	304	79-32	L	5/8/79	O	1.2.b.2		Plunger in UV coil out of adjustment.	W
Zion 2	304	79-49	L	10/9/79	O	1.2.b.2		Binding of UV coil mechanism trip lever.	W

* Not available.

\+ Submitted by licensee by NPRDS report only..

++ Submitted by licensee by both LER and NPRDS report.

Notes to Comments Column

(1) Where no comments are given for the items listed, only a single failure was involved. Based on this, and the specific comments given (together with any associated notes), there is a total of 54 failures for the 43 reports listed. Twenty different plants were involved.

(2) Involved one breaker in one testing session. Considered to be one failure for statistical purposes.

(3) Considered to be three failures for statistical purposes since the events on 1/31/79 involved one breaker in one testing session.

(4) Considered to be one failure for statistical purposes. The testing on 12/26/73 was performed with the UV trip mechanism deliberately degraded.

(5) Considered to be one failure for statistical purposes since the additional failure occurred during trouble shooting of the problem.

Appendix C

Variations Noted in Licensees' Bases for Reporting the Reactor Trip Breaker Failures Included in Appendix B

(1) Reported to NRC by Information Letters

As discussed in more detail in Section 2.1.4.1, the licensee for Robinson Unit 2 reported two events, occurring on 9/23/81 and on 12/20/82, to the NRC as information letters rather than LERs.

(2) Reported on Basis of Paragraph 1

The letters forwarding the Davis Besse and Kewaunee LERs referenced only the general heading of "Reporting Requirements" (Appendix A, Paragraph 1) as the basis for reporting the events without identifying which specific subparagraphs of Paragraph 1 where applicable.

(3) Reported on Basis of Paragraph 1.2

The forwarding letters for two of the Zion Unit 1 LERs referenced only the general introductory paragraph of "Reportable Occurrences" (Appendix A, Paragraph 1.2) as the basis of reporting without identifying which specific subparagraphs of Paragraph 1.2 were applicable.

(4) Reported on Basis of Paragraph 1.2.a

The forwarding letters for two Arkansas Unit 1 LERs and one Robinson Unit 2 LER referenced only the general introductory paragraph of "Prompt Notification with Written Followup" (Appendix A, Paragraph 1.2.a) as the basis of reporting without identifying which specific subparagraphs of Paragraph 1.2.a were applicable. For the Arkansas events, one occurred during testing and was reported as a 14-day report, while the other occurred on demand during reactor trip and was reported as a 30-day report.

(5) Reported on Basis of Paragraph 1.2.a.9

The forwarding letter for the one St. Lucie LER cited Appendix A, Paragraph 1.2.a.9.

(6) Reported on Basis of Paragraph 1.2.b

The forwarding letter for one Arkansas Unit 1 LER only referenced the general introductory paragraph of "Thirty Day Written Reports" (Appendix A, Paragraph 1.2.b) as the basis of reporting without identifying which specific subparagraph of Paragraph 1.2.b was applicable.

(7) Reported on Basis of Paragraph 1.2.b.1

The forwarding letters for three LERs cited Appendix A, Paragraph 1.2.b.1 as the basis for reporting.

(8) Reported on Basis of Paragraph 1.2.b.2

The forwarding letters for 17 LERs cited Appendix A, Paragraph 1.2.b.2 as the basis for reporting. Of these 17 30-day LERs, 13 involved RTB failure during surveillance testing while the other four (the 1/6/83 event at Salem Unit 2 and all three Zion Unit 2 LERs) involved RTB failure on demand during reactor trip. In regard to the 3/27/77 event at Zion Unit 2, the licensee's forwarding letter states that, "this event was previously classified as non-reportable but, as the result of an internal audit was reclassified as a 30-day report on 6/7/79." Accordingly, the licensee submitted the 1977 report on 6/7/79.

Appendix D

10 CFR 50.72 Requirements

Part 50.72 Notification of significant events.

(a) Each licensee of a nuclear power reactor licensed under §50.21 or §50.22 of this part shall notify the NRC Operations Center as soon as possible and in all cases within one hour by telephone of the occurrence of any of the following significant events and shall identify that event as being reported pursuant to this section:

(1) Any event requiring initiation of the licensee's emergency plan or any section of that plan.

(2) The exceeding of any Technical Specification Safety Limit.

(3) Any event that results in the nuclear power plant not being in a controlled or expected condition while operating or shut down.

(4) Any act that threatens the safety of the nuclear power plant or site personnel, or the security of special nuclear material, including instances of sabotage or attempted sabotage.

(5) Any event requiring initiation of shutdown of the nuclear power plant in accordance with Technical Specification Limiting Conditions for Operation.

(6) Personnel error or procedural inadequacy which, during normal operations, anticipated operational occurrences, or accident conditions, prevents or could prevent, by itself, the fulfillment of the safety function of those structures, systems and components important to safety that are needed to (i) shut down the reactor safely and maintain it in a safe shutdown condition, or (ii) remove residual

heat following reactor shutdown, or (iii) limit the release of radioactive material to acceptable levels or reduce the potential for such release.

(7) Any event resulting in manual or automatic actuation of Engineered Safety Features, including the Reactor Protection System.

(8) Any accidental, unplanned, or uncontrolled radioactive release. (Normal or expected releases from maintenance or other operational activities are not included.)

(9) Any fatality or serious injury occurring on the site and requiring transport to an offsite medical facility for treatment.

(10) Any serious personnel radioactive contamination requiring extensive onsite decontamination or outside assistance.

(11) Any event meeting the criteria of 10 CFR 20.403 for notification.

(12) Strikes of operating employees or security guards, or honoring of picket lines by these employees.

(b) With respect to the events reported under paragraphs (a)(1), (2), (3), and (4) of this section, each licensee, in addition to prompt telephone notification, shall also establish and maintain an open, continuous communication channel with the NRC Operations Center, and shall close this channel only when notified by the NRC.

(Sec. 161b. and o., Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[45 FR 13435, Feb. 29, 1980]

Appendix E

History of Major Actions Taken on RTB Failures

The AEC issued IE Bulletin No. 71-2 on December 9, 1971, as a result of the three failures of Westinghouse DB type breakers at Robinson Unit 2 on May 14, 1971, and two failures of similar breakers at Haddam Neck (Connecticut Yankee) on December 2, 1971. The Bulletin informed operating PWR licensees of the RTB failures and requested information on the results of testing, inspections, and corrective actions taken, or planned, by other facilities using similar RTBs. Following the further failures at Robinson Unit 2 on December 21, 1973, Westinghouse issued Technical Bulletin NSP-TB-74-1 (on January 11, 1974), and NSD Data Letter 74-2 (on February 19, 1974), regarding the requirements for inspection/maintenance of DB type breakers.

Because of the failures of General Electric AK type reactor breakers at Arkansas Unit 1, Crystal River Unit 3, Oconee Units 1 and 3, and Three Mile Island Unit 1, between April 25, 1975 and January 31, 1979 (and additional failures of similar breakers used in other safety related applications), the NRC issued IE Bulletin No. 79-09 on April 17, 1979 to all nuclear power plant licensees. The Bulletin also included General Electric Service Advice Letter No. 175 (CPDD) 9.3 regarding inspection/maintenance of AK type breakers.

Further, due to the November 30, 1980 breaker event resulting from an adjustment problem at St. Lucie, the NRC issued IE Circular No. 81-12 on July 22, 1981 recommending that licensees independently test the UV trip device and the shunt trip device during RTB surveillance testing.

In addition, due to the two ATWS events in February 1983 involving concurrence RTB failures at Salem Unit 1 (which uses Westinghouse DB type breakers), and the March 1983 breaker failures during testing at San Onofre Units 2 and 3 (which use General Electric AK type breakers), the NRC issued IE Bulletin No. 83-01 (on February 25, 1983), and Bulletin No. 83-04 (on March 11, 1983),

respectively. On April 1, 1983, the NRC issued IE Information Notice No. 83-18 which described recent failures (that occurred during the testing required in response to IE Bulletins No. 83-01 and No. 83-04) of reactor trip system circuit breakers with UV trip attachments; the Notice also provided additional information related to the UV attachment.

Since February 1983, considerable attention has been focused on RTB design, reliability, and performance history by the NRC, licensees, manufacturers, NSSS, INPO and others. Various studies and reports have been prepared addressing the Salem ATWS events themselves the generic implications of the events and the lessons to be learned.

Appendix F

Summary of RTB Failures Reported to NRC
Since the Salem Events (as of April 30, 1983)

Plant	Docket 50-	Date of Event	Licensee Description	Licensee Reasons for Not Previously Reporting
(Failures Prior to Salem Unit 1 Events)				
Arkansas-1* (GE AK-2 type breakers)	313	8/23/82	"B" RTB failed to trip open on demand.	Control rods were inserted in core with reactor subcritical at the time of occurrence.
Arkansas-1* (GE AK-2 type breakers)	313	10/6/82	"B" RTB failed to trip on demand from the non-safety related shunt coil, (during test).	Safety related portion (of breaker) required to be operable had functioned properly.
Arkansas-2* (GE AK-2 type breakers)	368	12/14/79	A new UV trip coil was installed in a RTB; reason for replacement is no longer known.	Occurred on a spare breaker (not in service) during refurbishment.
Arkansas-2* (GE AK-2 type breakers)	368	9/11/81	RTB TCB-5 would not stay closed.	Breaker tripped, thus performing its designed safety function and meeting T.S. Operability requirements.
Arkansas-2* (GE AK-2 type breakers)	368	11/12/82	New UV trip mechanism installed on a RTB during replacement of a blown control power fuse; reason for replacement of UV mechanism is no longer known.	Control rods were all inserted in the core and the reactor was subcritical. Also, RTB tripped properly on demand, thus meeting T.S. Operability requirements.
Calvert Cliffs-1 (GE AK-2 type breakers)	317	3/1/78	(TCB-2) U/V device failed to trip the breaker during performance of preventive maintenance.	Occurred during preventive maintenance during refueling conditions, with the unit in a shutdown mode.
Calvert Cliffs-2 (GE AK-2 type breakers)	318	12/20/82	(TCB-7) U/V device failed to trip the breaker during performance of preventive maintenance.	Occurred during preventive maintenance during refueling conditions, with the unit in a shutdown mode.

* Licensee states that ANO-1 and 2 Technical Specifications require reporting of a RTB failing to trip on demand during modes of plant operation when the RPS is required to be operable.

Also, the licensee states that ANO-1 (B&W plant) uses a safety related UV trip mechanism and a non-safety related shunt trip mechanism. However, both of the trip mechanisms for ANO-2 (CE plant) are safety related.

Plant	Docket 50-	Date of Event	Licensee Description	Licensee Reasons for Not Reporting
Calvert Cliffs-2 (GE AK-2 type breakers)	318	12/20/82	(TCB-1 & TCB-4) U/V devices exhibited sluggish operation during performance of preventive maintenance.	Occurred during preventive maintenance during refueling conditions, with the unit in a shutdown mode.
McGuire-2 (Westinghouse DS type breakers)	370	Early 1983	During testing; five failures of one RTB to trip on an UV signal were identified.	Not specifically stated. (Apparently because plant was still in pre-operational status. Plant received operating license on March 3, 1983).
Oconee-3 (GE AK-2 type breakers)	287	12/17/78	During startup procedures, one breaker failed to open when tested.	At the time of the event, it was not considered reportable by the licensee. The licensee indicated that failures of these breakers to function would be considered reportable pursuant to existing technical specifications.
San Onofre-2 (GE AK-2 breakers)	361	3/25/82	During surveillance four RTBs failed to trip on UV.	Personnel error. Submitted by LER 82-175/1T-0 on 3/25/82.
San Onofre-2 (GE AK-2 type breakers)	361	July 1982	In early July 1982, three RTBs failed to latch. On July 12, 1982, UV trip mechanism not operable on two RTBs.	Personnel error. Submitted by prompt report letter dated 3/30/83 and supplemented by LER 82-176//1T-0 on 4/12/83.
Three Mile Island-1 (GE AK-2 type breakers)	289	11/19/76	One trip breaker failed to trip during Post Maintenance Testing. Reactor was in a shutdown condition and other CRD breakers were racked out.	Binding of the UV device was caused by transporting and installing the breaker and the malfunction was detected before returning the system to service.
(Failures After the Salem Unit 1 Events.)				
Arkansas-1 (GE AK-2 type breakers)	313	3/23/83	No failures occurred during the Bulletin tests; however, while performing other tests on 3/23/83 the "A" RTB failed to trip when the UV device was de-energized from the control room.	

Plant	Docket 50-	Date of Event	Licensee Description
Calvert Cliffs-1 & 2 (GE AK-2 type breakers)	317 318	3/16/83	Four breakers in Unit 1 and three breakers in Unit 2 operated sluggishly.
Farley-2 (Westinghouse DS type breakers)*	364	4/14/83	During bench test, one RTB failed to open. The RTB contained a new, modified UV trip attachment.
Maine Yankee (GE AK-2 type breakers)	309	3/15/83	Three breakers operated sluggishly.
McGuire 1 & 2 (Westinghouse DS type breakers)*	369 370	3/18/83 and 4/2/80	No breaker failures occurred during the tests required by the Bulletin; however, on March 18 and 19, 1983 a reactor trip breaker failed to open 5 times at Unit 2 and another RTB failed to open twice at Unit 1. On April 2, at Unit 2 one RTB failed to function on the first operation after installation due to dislocation of the roller arm shaft of the UV device because the shaft retaining clip ring was missing.
Salem-1 (Westinghouse DB type breakers)	272	4/25/83	One RTB would not latch in closed position (coil spring was unattached).

* On April 20, 1983, Westinghouse Product Division notified the NRC by telecon of a Part 21 report of problems with their DS-416 series RTBs concerning a large number of UV trip attachments that did not conform to design dimensions and several had retaining ring seating defects which could impede proper tripping of the associated RTB. Five operating reactors (Farley Units 1 and 2, McGuire Units 1 and 2, and Summer) and 24 units under construction could be affected. Westinghouse also presented recommended actions to justify continued operation. The NRC initiated safety reviews on the reliability of the DS-416 breakers as their reliability appears to be substantially lower than the Westinghouse DB series and General Electric AK-2 series breakers.

Plant	Docket 50-	Date of Event	Licensee Description
San Onofre-2 and 3 (GE AK-2 breakers)	361 362	3/1/83 3/8/83	Three breakers in Unit 2 (on 3/8/83) and one in Unit 3 (on 3/1/83) failed to open when the UV trip devices were tested.

Appendix G

Stem-And-Leaf Plots of
Failure Rates
Figures G.1 through G.3

EXPLANATION

To explore the distribution of failure rates among the individual plants, AEOD plotted the point estimates in a format called "stem-and-leaf." In these plots, each digit to the right of the vertical line holds the place for one failure rate. This digit is numerically the second digit of the failure rate. The first digit is taken from the left side of the vertical line. Examples are shown below.

```
          6 |
   -2       |                    -2
  10      7 | 3 - - - -7.3 x 10    or 0.073 RTB failures per breaker year.
          8 |
          9 |
------------------------------------------------------------
          1 | 4[6] - - -1.6 x 10^-1  or 0.16      [ ] - Group average RTB
   -1       |                                            failure rate per
  10      2 |  - - - -1.4 x 10^-1   or 0.14               breaker year.
          3 |
          4 |
          5 |
```

In all the plots, the exponent is indicated by 10^{-1} or 10^{-2} with a dotted horizontal line marking the transition.

The point estimate, 0.073 failures per breaker year, indicates that one breaker failure would be expected each 14 years on the average.

A comparison of the individual plants to the group average value indicates the distribution of the data, i.e., the individual plants do not cluster near the group average.

Figure G.1

RTB Failure Rate Per Breaker Year - B&W Plants

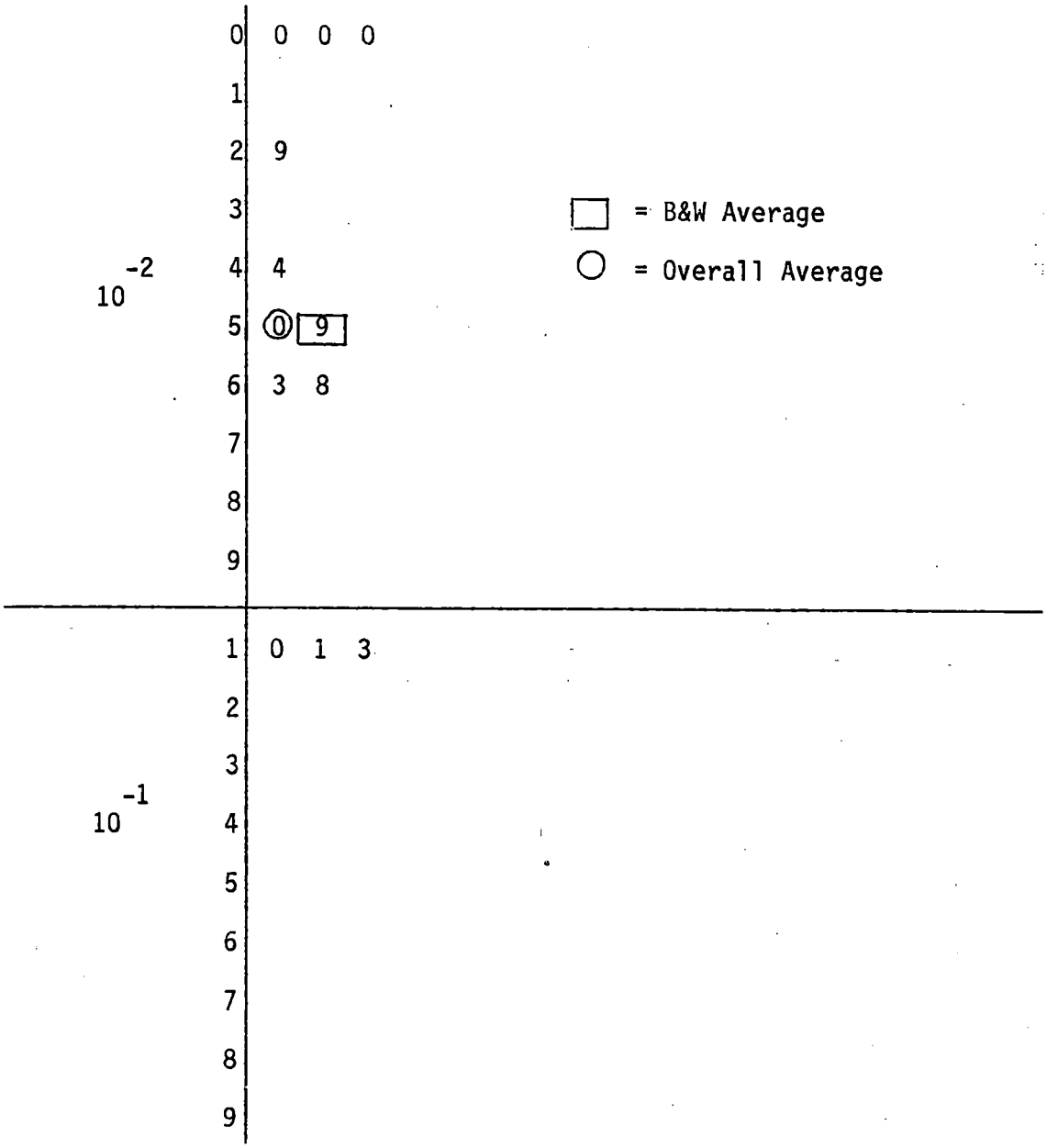


Figure G.2

RTB Failure Rate Per Breaker Year - Westinghouse Plants

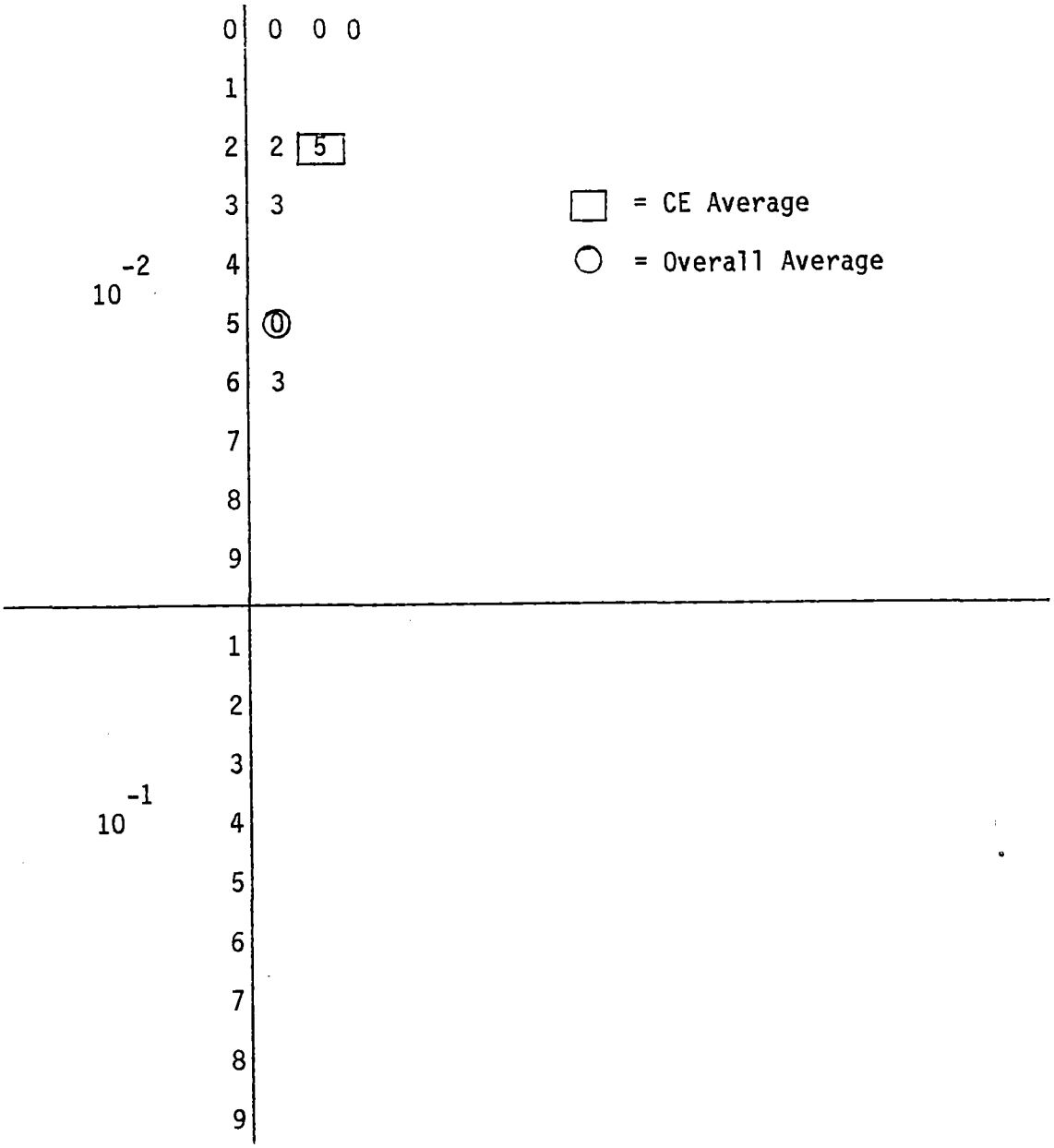
10^{-2}	0	0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0
	1	
	2	
	3	
	4	2
	5	3 6 0
	6	
	7	
	8	4
	9	
10^{-1}	1	1 5 9
	2	2 4
	3	0
	4	0
	5	
	6	
	7	
	8	
	9	

□ = W Average

○ = Overall Average

Figure G.3

RTB Failure Rate Per Breaker Year - CE Plants



Appendix H
Post TMI Requirements for Operating Reactors
(NUREG-0737; Item I.C.5)

PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

Position

In accordance with Task Action Plan I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660)," each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for the review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to assure that important information on operating experience originating both within and outside the organization is continually provided to operators and other personnel and that it is incorporated into plant operating procedures and training and retraining programs.

Those involved in the assessment of operating experience will review information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE Bulletins, Circulars, and Notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient importance that it must be

dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special changes to operating and emergency procedures, issuance of special precautions, etc.) and must be handled in such a manner to assure that operations management personnel would be directly involved in the process. In many other cases, however, important information will become available which should be brought to the attention of operators and other personnel for their general information to assure continued safe plant operation. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealt with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. It is important also that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information.

Applicability

This requirement applies to all operating reactor and applicants for operating license.

Implementation

Procedures governing feedback of operating experience to plant staff shall be completed and the procedures put into effect on or before January 1, 1981 or prior to issuance of an operating license, whichever is later.

Type of Review

A post-implementation review will be performed.

Documentation Required

No documentation is required.

Technical Specification Changes Required

Changes to technical specifications will not be required.

References

NUREG-0660, Item I.C.5

Letter from D. G. Eisenhut, NRC, to All Licensees, dated May 7, 1980.