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NUREG-1272

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Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data — 1986

**U.S. Nuclear Regulatory
Commission**

Office for Analysis and Evaluation of Operational Data

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Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data — 1986

Manuscript Completed: April 1987
Date Published: May 1987

NUREG--1272-Vol. 1

TI87 900765

Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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ABSTRACT

This annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during calendar year 1986. Comments and observations are provided on operating experience at nuclear power plants and other NRC licensees, including results from selected AEOD studies; summaries of abnormal occurrences involving U.S. nuclear plants; reviews of licensee event reports and their quality, reactor scram experience from 1984 to 1986, engineered safety features actuations, and the trends and patterns analysis program; and assessments of nonreactor and medical misadministration events. In addition, the report provides the year-end status of all recommendations included in AEOD studies, and listings of all AEOD reports issued from 1980 through 1986.

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OFFICE FOR ANALYSIS AND EVALUATION OF OPERATIONAL DATA

ANNUAL REPORT

1986

1. INTRODUCTION

NRC's Office for Analysis and Evaluation of Operational Data (AEOD) was established in 1979, as one of the Commission's earliest major steps toward improving the use of licensee operating experience to identify and resolve problems with potential safety-related implications. The Office, which reports directly to the Executive Director for Operations, is dedicated to the collection, assessment, and feedback of operational data.

AEOD's focus and role in the regulatory program is to provide a strong capability for the analysis of operating experience, independent of the routine regulatory activities associated with licensing, inspection, or enforcement and to feed back the lessons learned to the NRC, the nuclear industry and the public.

AEOD accomplishes its mission through analysis and evaluation of operational safety data associated with all NRC-licensed activities. These include the operations of commercial power reactor licensees and radioactive material and fuel cycle licensees. The Office also coordinates the overall NRC operational data program, and serves as the focal point for interaction with outside and foreign organizations performing similar work. The Office's objectives and its major tasks and activities are highlighted below.

Objectives

- ° Collect, screen, analyze, and feed back operating experience to appropriate NRC offices, the nuclear community (domestic and international), and the public for all NRC-licensed activities.
- ° Coordinate the overall NRC operational data program.
- ° Administer the NRC's Incident Investigation Program.

Specific Tasks and Activities

- ° Screen U.S. and foreign operational events for significance.
- ° Systematically and independently analyze operational events and issue technical reports on the results.
- ° Seek trends and patterns in operating experience which may indicate potential safety problems.

- ° Develop the procedures for and support the establishment of Incident Investigation Teams.
- ° Develop and track AEOD recommendations for action by other NRC Offices for resolution of safety issues.
- ° Develop and coordinate operational data retrieval systems, including foreign data.
- ° Prepare and coordinate Abnormal Occurrence reports to Congress and the public.
- ° Prepare Power Reactor Events reports and other feedback documents.
- ° Provide documentation of U.S. events for reporting to the Nuclear Energy Agency's Incident Reporting System.
- ° Serve as principal point of contact with ACRS, INPO, and NSAC on operational data activities.

AEOD maintains an awareness of the studies undertaken by other organizations such as the NRC Offices of Nuclear Reactor Regulation (NRR) and Inspection and Enforcement (IE), and the Institute of Nuclear Power Operations (INPO). Normally, AEOD will not overlap or duplicate the study efforts of other organizations unless a particular need or special circumstance exists. Thus, AEOD does not review in-depth all events or operating problems because many problems identified through operating experience are being studied extensively and comprehensively by other organizations. Thus, to be aware of all identified safety problems and lessons of experience, one must integrate the output of many NRC offices and outside programs.

The recommendations contained in AEOD studies discussed in the following sections are not final NRC positions. They are internal recommendations for action by appropriate NRC program offices (e.g., NRR, IE) or Regional Offices. The program office or Regional Office is responsible for reviewing and, where appropriate, implementing AEOD recommendations. A written response to each recommendation is required and a formal action tracking system has been established.

2. ORGANIZATION AND STAFFING

At the close of CY 1986, major changes in the scope of the Office's objectives and activities were announced as part of the pending NRC reorganization (April 1987). Following this reorganization, the staff will more than double, and additional responsibilities will include developing and managing the NRC program for emergency and incident response, developing and providing technical training for NRC staff, developing and managing the NRC program for reactor performance indicators, and managing and conducting the support functions for the Committee to Review Generic Requirements. AEOD's key duties of analyzing and evaluating operational safety data associated with NRC-licensed activities, independently identifying issues that require action, and feeding back those findings to the appropriate recipients (other NRC offices, licensees, the industry, Congress, etc.) are thus enhanced. (For details, see NUREG-0325, Rev. 10, issued February 1987.)

During 1986, a number of changes were made in AEOD management. Vacancies in the positions of Chief of the Reactor Operations Analysis Branch (ROAB), Chief of the Program Technology Branch (PTB), and Chief of Reactor Systems Section 1 in ROAB were filled. Also during 1986, AEOD recruited for four Reactor Systems Engineers for vacancies in ROAB, and one Plant Systems Engineer and one Reactor Systems Engineer were added to PTB. One of these individuals is a Senior Reactor Operator-licensed engineer and two are former resident inspectors.

AEOD's staffing level for FY 87 (excluding the pending reorganization) is essentially at full strength (43 FTE), and thus remains at approximately the same level as in FY 86. The AEOD organizational structure and staffing as of December 31, 1986 are shown below.

3. COMMENTS AND OBSERVATIONS ON NUCLEAR POWER PLANT OPERATING EXPERIENCE

AEOD activities include studies to identify and evaluate potentially significant events and safety concerns involving U.S. commercial power reactors, based on events reported to the NRC by nuclear power plant licensees. During 1986, these events were reviewed individually as well as collectively by AEOD as part of its engineering assessment and trends and patterns activities. Studies were conducted that provide a broad overview of the operating experience and characteristics of the nuclear industry as well as individual facilities. This section provides some observations and perspectives on selected aspects of operating experience analyzed in 1986, and provides overviews of Licensee Event Reports as well as the significant events forwarded to the Commission for consideration as Abnormal Occurrences.

3.1 Overview of Operating Experience

In the review of operating experience, certain common characteristics in a number of areas and a number of potentially generic issues were noted which may warrant increased NRC and industry attention. These are discussed below.

3.1.1 Review of Selected Significant Events

AEOD has a study in progress of significant events that occurred at nuclear power plants in the period from January 1985 to July 1986. The objectives of this study are to determine what common characteristics the events shared, and to assess the additional lessons to be learned from the collection of events. Events are included in the study if they met one or more of the following criteria:

- ° The event was studied by an Incident Investigation Team (IIT) or an Augmented Inspection Team (AIT).
- ° The event was determined to be an abnormal occurrence.
- ° The event had a calculated conditional probability of core damage of 1×10^{-4} or greater (see Section 5.2.6.1).

A total of 21 events met one or more of the selection criteria. Sixteen of the 21 events were operating events, four were events involving latent defects, and one event involved premature criticality during startup. Table 1 lists the events considered in this study.

The 21 events in the study included a range of situations. Many events represent scrams or trips that were accompanied by equipment failures. None of the 21 events resulted in significant plant damage, or caused a significant increase in the amount of radioactive material released from the plant. Furthermore, with a few exceptions, the significant events included in the review did not generally represent severe challenges to plant equipment or the operating staff.

Table 1
Significant Events Meeting Selection Criteria
January 1985 through June 1986

REACTOR	LER NO.	DATE	IIT	AIT	AO	CDP
Operating Events						
DAVIS-BESSE	34685013	6/9/85	X		X	X
SAN ONOFRE 1	20685017	11/21/85	X		X	X
RANCHO SECO 1	31285025	12/26/85	X		X	
D. C. COOK 2	31685035	10/29/85		X	X*	
PILGRIM 1	29386009	4/12/86		X		
PALISADES	25586018	5/19/86		X		
LASALLE 2	37486011	6/1/86		X	X	
CATAWBA 2	41486028	6/27/86		X	X	
TURKEY POINT 3	25085021	7/22/85				X
HATCH 1	32185018	5/15/85				X
TROJAN	34485009	7/20/85				X
DAVIS-BESSE	34685002	1/15/85				X
OYSTER CREEK	21985012	6/12/85				X
HATCH 1	32185010	1/16/85				X
GRAND GULF 1	41685050	12/31/85				X
HATCH 2	36685030	11/5/85				X
Events Involving Latent Defects						
CALVERT CLIFFS 1	N/A	7/85		X	X*	
MAINE YANKEE	30985009	8/7/85			X	
BRUNSWICK 2	32585008	9/27/85			X	
PILGRIM	N/A	5/19/86			X	
Startup Event						
SUMMER	39585003	2/28/85			X	

IIT = Event was subject of incident investigation
 AIT = Event was subject of augmented inspection
 AO = Event was determined to be an abnormal occurrence
 CDP = Conditional core damage probability estimated to be greater than 1×10^{-4}

*Update of AO prior to 1985.

The characteristics of the events can be divided into standard subsets: hardware (e.g., design, common mode failures) and human factor deficiencies. Table 2 lists the types of deficiencies identified with each of the events.

Table 2
Event Characteristics

Significant Event	Date	Design Problems	Deficiencies					
			Hardware		Generic Problems	Human Factors		
			Common Mode Failures	System Inter-action		Procedure	Training	
Events Involving Transients								
DAVIS-BESSE	6/9/85	X	X		X	X	X	
SAN ONOFRE 1	11/21/85	X	X	X	X	X	X	
RANCHO SECO 1	12/26/85	X	X	X	X	X	X	
D. C. COOK 2	10/29/85	X	X	X				
PILGRIM	4/12/86	X	X		X	X	X	
PALISADES	5/19/86	X			X	X	X	
LASALLE 2	6/1/86	X	X	X		X		
CATAWBA 2	6/27/86	X	X	X		X	X	
TURKEY POINT 3	7/22/85	X	X	X	X	X	X	
HATCH 1	5/15/85	X	X	X	X	X		
TROJAN	7/20/85	X	X		X	X		
DAVIS-BESSE	1/15/85	X	X				X	
OYSTER CREEK	6/12/85	X				X		
HATCH 1	1/16/85	X						
GRAND GULF 1	12/31/85	X	X	X				
HATCH 2	11/5/85	X		X				
Events Involving Latent Defects								
CALVERT CLIFFS 1	7/85	X	X	X		X		
MAINE YANKEE	8/7/85		X			X		
BRUNSWICK 2	9/27/85	X	X	X	X			
PILGRIM	5/19/86	X	X					
Startup Event								
SUMMER	2/28/85					X	X	

3.1.1.1 Hardware

It was found that 20 of the 21 events involved one or more hardware problems. In general, these could be divided into four categories, although the categories are not independent: design problems, common mode failures, system interaction problems, and generic problems. Some examples of these included:

- ° Design Problems
 - Design did not meet requirements (e.g., a residual heat removal system was vulnerable to single failures; an air system did not produce the quality of air required for the associated components to perform as intended).
 - Inadequate instrumentation and control misled or confused operators (e.g., a lack of indication of the status of steam generator blow-down on control panels).
 - Failure to understand and account for component behavior under actual operating conditions (e.g., incorrect specification of initial torque switch bypass settings; check valves specified for use in flow conditions under which they will be damaged, such as high turbulence for extended periods of time).
- ° Common Mode Failures
 - Failure of a number of motor operated valves to open for the same reason (e.g., inadequate torque switch bypass switch settings).
 - Potential failure of multiple trains as the result of procedure/design problems (e.g., residual heat removal system was susceptible to single failure; multiple pressure sensing lines were unavailable due to an inadequate procedure).
 - Failures ascribed to degradation of identical equipment (e.g., main steam isolation valve; reactor trip breaker; Static "0" Ring switches).
- ° System Interactions Problems
 - Interaction between nonsafety air system and safety systems.
 - Interaction between nonsafety integrated control system and safety systems.
 - Interaction between systems caused by electrical noise.
- ° Generic Problems
 - In many of the events, the deficiency or failure could be generic in nature (i.e., identical components used in other nuclear power plants might be subject to the same type of failure).

It can be seen (Table 2) that many of the events involved several different types of hardware problems. Design problems were seen in 19 of the 21 events; potential or actual common mode failures in 16 of 21; generic problems in 12; and system interactions in nine of the events. In some of the events, the hardware problem occurred as the result of plant modifications; in others, the event stemmed from the original plant design. Essentially all of the operating events (13 out of 16) occurred at plants that had been operating for 7 or more years. All of the IIT events, and three of the five AIT events occurred at plants that had been operating for 10 years or more. This finding points out that not all hardware problems have been uncovered and corrected early in plant life.

3.1.1.2 Human Factors

The dominant human factors problems found in the 21 events were:

- ° Procedures - Procedural deficiencies, use of ad hoc procedures, deviations from procedures, or lack of procedures were contributing factors to many of the 21 events. Some examples include:
 - Failure of a procedure to give clear direction on when to take a specific action.
 - Inservice testing procedures that were inadequate to detect a damaged or degraded valve.
- ° Training - Training was identified as a problem in a number of the 21 events, especially in those events that were the subject of an IIT or an AIT investigation. Some examples of training deficiencies included: failure of operator training to focus on normal operations and operational events; training for complex calibration or maintenance activities; and operator training on a simulator that did not sufficiently represent the actual plant.

As noted above, none of these 21 events resulted in significant plant damage, and few of the events presented severe challenges to plant equipment or operating staff. However, these events (especially the operating events) illustrate events or sequences of events that, under different circumstances, could result in challenges to plant equipment or operators. Generally, the operating events were initiated because of some random component failure or human error, frequently in a non-safety system. In most cases, the reactor tripped or scrammed correctly (automatically or manually) following the initial difficulty. However, often in the case of significant operating events, there were additional system difficulties that occurred following the trip or scram. These difficulties resulted from a variety of reasons, but rarely from random equipment or component failure. In several cases, the system difficulties arose from a latent defect such as a potential common cause failure of redundant equipment or an unexpected system interaction or a failure of personnel to understand the system design and component behavior. In some events, human error, arising because of procedural deficiencies or training deficiencies, contributed to the extent of post-trip problems, or to the duration of a problem.

The preliminary findings from the review of these significant events are that most events involved more than one contributor; i.e., hardware and human factors problems. The many design problems and common mode/generic failures in the events are of concern, particularly since some of the equipment deficiencies experienced in these events were not identified by previous design reviews and testing. The preliminary review of these 21 events showed that most events involved a reactor trip or scram complicated by safety system losses after the event. In some cases the safety system loss was attributable to a lack of understanding of how system operating conditions could affect the operation of a component. To reduce the occurrence of the type of significant events reviewed in this report, efforts should be directed to reducing scrams or trips. In addition, efforts to reduce or eliminate safety system failures should include, as an initial step, a process designed to develop an understanding of the root cause of the safety system failure, and to correcting the root cause.

3.1.2 Common Mode Failure Mechanisms Continue to be Identified through In-Depth Studies of Operational Events

Since 1980, AEOD has evaluated operational data with a primary goal of finding and feeding back information on safety significant situations which, if left uncorrected, could result in potentially serious operating events. AEOD has found many such situations, a number of which involved construction deficiencies, system interactions, and common mode failure mechanisms that eluded plant safety analysis reviews. Upon discovery of such deficiencies and situations, AEOD has alerted other NRC program offices and industry groups of their existence, potential significance, and the need for corrective action.

The potential for common mode failure represents a latent problem that may pose a significant challenge to plant safety following a major transient or accident. Such failures normally should be identified through detailed design reviews and related test programs, yet a number of potential common mode failures continue to be identified after plants begin commercial operation. The potential common mode failures found and the AEOD reports* which documented the findings to alert the NRC program offices and industry include:

- ° BWR scram system failures (case studies C001, C002, C103, and C403; engineering evaluations E007, E015, and E018)
- ° Air system failures [case study C204 and a draft case study C701 (see Section 3.2.3); engineering evaluation E123]
- ° Instrumentation and control system and electrical equipment failures (case studies C402 and C604)
- ° Service water system failures (case study C202; engineering evaluations E016 and E215)
- ° Motor operator valve failures (case studies C203 and C603; special study S501)
- ° Auxiliary feedwater system failures (case studies C404 and C602; engineering evaluation E325)

*Listings of AEOD reports issued from 1980 to the present are provided in Appendix B.

- ° Diesel generator failures (engineering evaluation E110; technical review T602)
- ° Gate valve failures (special study S402)
- ° Aging of batteries (engineering evaluation E307)
- ° Check valve failures affecting emergency core cooling systems (case study C502)
- ° Failure of intermediate head emergency core cooling system pumps (special study S603)
- ° Erosion of water and steam lines (engineering evaluation E416)

These potential common mode failures exemplify the fact that it is necessary to carefully review operating experience and to identify and understand the exact cause of failure. Otherwise the corrective action is not likely to be effective and the probability of recurrence may not be reduced. The discovery of root causes, however, is often difficult and can involve a wide range of in-depth studies of operating events. The identification of common mode failures, such as those above, continues to be important, since as discussed in Section 3.1.1, serious operating events typically involve a combination of hardware and human performance problems.

3.1.3 Repeated Failures May Indicate Incorrect Identification of Root Cause, or Ineffective Corrective Actions

AEOD case studies in 1986 addressing failures of inverters, motor operated valves, and electronic components in instrumentation and control systems identified recurring failures of this equipment at operating nuclear plants. While the reasons for these recurring failures cannot be determined with certainty, such failures seem to reflect incorrect or incomplete root cause determinations, or a lack of effective corrective actions.

In the AEOD case study on "Operational Experience Involving Losses of Electrical Inverters," discussed in Section 3.2.5, it is concluded that the number of inverter losses per reactor-year shows little or no improvement in each calendar year in the 3 years (1982, 1983 and 1984) included in the study. The overall lack of improvement has occurred despite numerous NRC and industry feedback communications on this problem and the recommendations for corrective actions. The review indicates that a major contributing factor for inverter failures is an incompatibility between the actual plant service conditions and the design service conditions (i.e., actual plant service conditions are more severe than those assumed in the design of inverters). The report identifies three potential failure mechanisms for inverters: (1) high ambient temperature and/or humidity within inverter enclosures; (2) electrical interconnections and physical arrangements of components which form the inverter circuitry; and (3) voltage spikes and perturbations at inverter inputs and outputs in conjunction with relatively rapid response time of solid state devices. However, effective corrective action depends on proper identification of the precise underlying failure mechanism, and this appears to be lacking.

The AEOD case study on "A Review of Motor Operated Valve Performance," discussed in Section 3.2.4, represented an extension and update of previous NRC studies and reports of motor-operated valve (MOV) operating experience. This study indicates that recent MOV events (from 1981 to mid-1986) involve failures that are similar to those observed earlier, and there is no apparent improvement in the rate of failure. Thus, the report concludes that the previous recommendations are still valid. Most of the valve inoperability was associated with items such as torque switch/limit switch settings, adjustments, or failures; motor burnout; improper sizing or use of thermal overload devices; premature degradation related to inadequate protective devices; damage due to misuse (valve throttling, and hammering of valve operator); mechanical problems (loosened part or improper assembly); or the bypass circuit around the torque switch not being installed or being improperly set. In many events, however, the true root causes of failures to operate were not adequately determined. The most important conclusion from this study concerning valve assembly operability and performance/reliability is that current methods and procedures at many operating plants require improvement to assure that MOV assemblies will operate when needed (e.g., under credible accident conditions). The study also concludes that assurance of valve operability appears to be strongly dependent upon the capability to correctly diagnose, assess, and evaluate valve assembly operational failures so that root causes of failure, including erroneous switch setpoints, are correctly determined and proper changes are implemented.

The AEOD case study on the "Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems," discussed in Section 3.2.6, documents the review and evaluation of four events involving failures of solid state electronic components due to overheating. One of the findings of the study was that identifying elevated room ambient temperatures or instrument cabinet temperatures as the root cause for the failure of electronic components generally has not been easy or immediate. Licensees, over an extended period, experienced several failures and many corrective actions before finally identifying overheating of components as the underlying reason for many of the failures they had experienced. It is also noted in the report that although several other operating plants have experienced failures of electronic components similar to those reviewed in the study, high ambient or internal cabinet temperature was not always pursued as a possible root cause of the failures. The report notes that overheating of electronic components raises two concerns that are generic to all operating nuclear plants that utilize heat sensitive electronic components: (1) decreased reliability of electronic equipment due to increased failure rate of components, and (2) the potential for common cause failure of safety-related instrumentation channels if there is extended loss of normal cooling air flow to the cabinets in which the instruments are located.

3.2 Results from Selected AEOD Studies

In 1986 a number of AEOD studies were completed or progressed to the point where conclusions could be formed. Several of the significant reports are highlighted in this section. Discussed below are the AEOD: (1) review of new plant operating experience; (2) study on the effectiveness of licensees' and the NRC's operational experience feedback programs; (3) case study on air system problems and their safety implications; (4) case study on motor-operated valve performance; (5) case study on operational experience involving electrical inverters; (6) case study on the effects of elevated room temperature on electronic components; and (7) case study on operational experience involving turbine overspeed trips.

3.2.1 Review of Operational Experiences at Recently Licensed (New) Plants

Among the concerns that arose from the TMI-2 accident, which occurred 13 months after licensing, was the adequacy of attention paid to operational safety by the regulators and plant operators during the initial phases of the plant's operation. The pace of licensing increased in the mid-1980s and, in this regard, both the 1985 and 1986 Policy and Planning Guidance of the Commission set goals to "...continue to closely monitor the first 2 years of operation of new plants coming on line." This led AEOD to conduct a study of the experience at new plants. Although the initial study, published in August 1986, identified significant differences in the experience of new plants, an analysis of the causes underlying the observed variation in event rates or the high frequencies was not performed. As a result, this study was expanded and continued to further identify the causes for these differences and the operational problems experienced by new plants. Three plants licensed in 1986 were added to the scope of the August report, and the plant event data (through June 1986) were examined. In order to take advantage of AEOD's Trends and Patterns Program for LER data, the event types covered in the analyses of this latter report are reactor scrams (a subset of reactor protection system actuations), engineered safety feature (ESF) actuations, violations of technical specifications and losses of system safety function.

This study includes 22 plants that received initial operating licenses after January 1, 1983, and reviews the reportable operational events which occurred within the first 24 months after OL issuance, up to a cutoff of June 30, 1986.

The analysis takes explicit account of the startup period and the plant status in the data for each plant. For example, Figure 1 shows scram counts displayed by months since OL. The months of initial criticality (IC), full power licensing (FP) and start of commercial operation (CO) are indicated, along with the reactor critical hours for each month and the cause category for each scram.

Figure 1. Unplanned Reactor Scrams

A formal statistical analysis using average unplanned event rates during both the startup program and the first 180 days after completion of the startup program showed no statistically significant correlation of these event rates with NSSS vendor, type of AE (the utility itself or an outside firm), the length of the startup program, date of OL, or prior nuclear plant experience. The analysis did indicate a significant correlation between the average rate during the startup program and the average rate during the first 180 days of commercial operation for scrams, ESF actuations and losses of system safety function. For violations of technical specifications, a significant correlation existed between the commercial rate and whether or not the plant was a second unit at a site (i.e., lower violation rates prevailed for subsequent units at a site).

The study of causes through a simple categorization process such as shown in Figure 1 too often masks the true nature of the events' root causes. Indeed, the information in LERs, even though subjected to a thorough review by licensees, often does not reflect underlying factors such as the overall management of operations or schedule pressures. Therefore, more detailed cause analysis was performed using startup reports, site discussions with licensees, discussions with NSSS vendors and other available documentation of early post-licensing operational experience. Site visits were conducted for the following plants: Diablo Canyon 1 and 2, Callaway, Byron 1, Wolf Creek, St. Lucie 2, Palo Verde 1 and 2, Washington Nuclear 2 (WNP-2), Susequehanna 2, and Hope Creek.

The preliminary findings and conclusions resulting from the continued AEOD study of new plants include the following:

- ° Some plants achieve commercial operation in a smooth manner. They have relatively few unanticipated events, similar to Japanese plants, which appear to be relatively event-free in their early years. Other domestic plants have difficult early years, and some may experience over 25 unanticipated scrams, over 100 unplanned actuations of plant safety features, and over 50 violations of their technical specifications during their first 2 years of operation.
- ° In general, newly licensed reactors experience a higher frequency of unplanned operational events during their early years of operation when compared to later years. This observation has historically been translated into segregating the statistics for this period from general industry figures, and has given rise to the views that (1) startup experience is less important than post-commercial operation, and (2) the high frequency of unplanned operational events is an inevitable and hence acceptable part of early operation. However, this study found that frequent difficulties are not necessarily inherent in the startup process.
- ° During this study, AEOD found that, individually, power reactor licensees already have recognized the need for action to achieve good early operation. They have developed programs in the areas of training and operations that cover the spectrum of actions which, if implemented effectively, would result in improvement.
- ° This study confirmed that it is possible to achieve significant reductions in the frequency of reportable events for new plants. The data analysis

of the operating experience provided a pointer to focus discussions with licensees and identified a number of lessons that could be used to achieve improvements.

- ° Both the NRC and industry have recently taken additional steps to focus on newly licensed plant operation, including the NRC senior management focus on this topic in October 1986 and the recently proposed INPO program for improving performance during early operation. The findings of this study reinforce the need for and benefit of these efforts.
- ° For a new plant, an aggressive root cause determination and corrective action program for responding to failures is a key ingredient to improvement. The analyses of operational data for scrams, engineered safety feature (ESF) actuations and loss of system safety function clearly demonstrate the need to correct the root causes for reportable events early in life. Without correction, these root causes will likely exist during the early commercial operation of the facility. At this time of life, the relatively high challenge frequency coupled with the potential of undetected systems problems may present a significant challenge to a new operating crew. Therefore, the analyses indicate that increased attention to operations is appropriate during the period of early commercial operation. The improvement measures identified in the report represent an effective action list to consider in formulating a program to prevent high reportable event frequencies.
- ° Nuclear safety measures consist of prevention and mitigation items. This study identified measures to reduce the frequency of reportable events, which are prevention initiatives. Mitigation measures such as operator training for response to plant transients are also necessary, since the two sets of measures do not overlap. However, it is necessary to carefully assess event reduction measures such as revisions of setpoints to avoid a net decrease in plant safety.

3.2.2 Operational Experience Feedback Programs

Subsequent to the TMI-2 accident, various studies highlighted a need for increased attention to the feedback of operating experience (OE) by the NRC, the industry, and each licensee in order to use the lessons of experience to prevent serious nuclear incidents from occurring. One action taken by the NRC was to require that each nuclear power plant licensee have a formal (i.e., controlled by procedures) program to incorporate the lessons learned from operating experience into plant hardware and procedures, and to assure that these lessons are communicated to plant operators and other personnel.

In fall 1984, AEOD initiated a survey to assess the usefulness of operational data feedback documents and to determine the characteristics of licensees' operating feedback programs. AEOD staff made on-site visits to seven licensees, met with INPO to discuss their programs, observed an INPO plant evaluation in the operational experience area, and in May 1986 issued AEOD report S602, "An Overview of Nuclear Power Plant Operating Experience Feedback Programs."

In general, the study found that, over the last 5 years, many significant and worthwhile initiatives have been implemented to understand the lessons of

experience. As a result, there has been increased attention paid to OE feedback. However, many licensee programs have not yet achieved effective performance levels in all areas. Most plants are making moderate, not extensive, use of their in-house operating experience, and in general are making less use of the large body of knowledge associated with events and concerns that originate elsewhere in the industry. Thus, many licensee OE review activities still do not provide high confidence that all of the important lessons of operating experience are being thoroughly explored and used. To increase the effectiveness of OE reviews in order to correct past and potential operational problems (i.e., assuring safety of operations), licensees and the industry, or the NRC need to take further actions.

Based on this study, AEOD recommended that IE consider informing licensees of the conclusions drawn from this study and suggesting that each licensee review their OE activities for effectiveness and conformance with the intent and scope of the existing NRC requirements. No major regulatory actions were recommended because AEOD was aware that industry initiatives in this area were still ongoing. AEOD recommended that the effectiveness of OE activities continue to be monitored by staff through operating experience reviews and observations of OE review activities. If, after sufficient time (2 years or so), the effectiveness of OE review activities has not improved, further regulatory actions should be considered. In this regard, AEOD recommended as a prudent measure that NRC initiate the development of better guidance for licensees' OE activities, and an inspection module based on that guidance. In addition, AEOD recommended that the staff consolidate some of the existing systems for communicating operating experience into a single "NRC Notice" system. This system would thus be the sole NRC system for the feedback of operating experience to licensees. Staff responses and the status of actions related to those recommendations are contained in Section 6.0.

3.2.3 Air Systems Problems at U.S. LWRs

A preliminary AEOD case study report on "Air Systems Problems at U.S. Light Water Reactors" was issued for peer review on December 11, 1986. The study analyzes and evaluates the operational experience related to, and the safety implications associated with, failures and degradations of air systems at U.S. LWRs.

The report presents aspects of air system degradations and plant responses to air system losses which are not addressed in previous studies. It also highlights more than two dozen events in which, contrary to licensing assumptions, a safety-related system failed due to an air system degradation or failure. Operating events involving the loss or degradation of air systems were judged to be safety significant because they may lead, under different circumstances, to potentially serious events and conditions which have not been analyzed in a plant's Final Safety Analysis Report (FSAR). Some of the systems which were significantly degraded or failed were decay heat removal, auxiliary feedwater, BWR scram, main steam isolation, salt water cooling, emergency diesel generator, containment isolation, and the fuel pool seal system.

The root causes of most of those failures were traceable to design and/or management deficiencies. The design and operating problems found appear to reflect a lack of sufficient regulatory requirements and review, and the view by many applicants and licensees that air systems are not highly important to

assuring plant safety. The report addresses specific deficiencies which were found in the following areas: (1) mismatched equipment - the air quality capability of the instrument air system filters and dryers does not always match the design requirements of the equipment using the air; (2) maintenance of instrument air systems is not always performed in accordance with manufacturer's recommendations; (3) air quality is not usually monitored periodically; (4) plant personnel frequently do not understand the potential consequences of degraded air systems; (5) operators are not well trained to respond to losses of instrument air, and the emergency operating procedures for such events are frequently inadequate; (6) at many plants the response of key equipment to a loss of instrument air has not been verified to be consistent with the FSAR; (7) safety-related backup accumulators do not necessarily undergo surveillance testing or monitoring to confirm their readiness; and (8) the size and the seismic capability of safety-related backup accumulators at several plants have been found to be inadequate.

This final case study was formally issued in March 1987.

3.2.4 A Review of Motor-Operated Valve Performance

AEOD issued case study C603 in response to Action Item 6(1) of the actions directed by the EDO to respond to the NRC staff investigation of the June 9, 1985 event at Davis-Besse. The report brings together previous NRC studies and other related documents (i.e., AEOD reports and IE bulletins, circulars, and information notices), reviews approximately 1200 valve operator events (found in the SCSS and NPRDS data bases; see Sections 5.2.2 and 5.2.7) from 1981 to mid-1985, and incorporates new data from an NRC valve testing program that utilized signature tracing techniques on valve assemblies in operating plants.

The report concludes that current methods and procedures at many operating plants require improvement to assure that valves will operate when needed. Furthermore, the deficiencies would generally not be detected by existing plant procedures intended to assure operability, such as surveillance testing (plant technical specifications and ASME Code, Section XI inservice testing) or plant operator observations. Thus, assurance of valve operability appears to be strongly dependent upon the diagnostic capability to assess and evaluate failures to operate so that root causes of failure--including switch setpoint--are correctly determined and proper changes implemented. Further, the issue of valve assembly performance and reliability is a complex subject that involves several technical disciplines.

AEOD currently is working with industry to develop and implement improved guidance, procedures, and/or equipment to address all aspects of safety-related motor-operated valve assembly operability. The overall goal for this industry effort is expected to involve the development of uniform guidance, procedures, and/or equipment which licensee management could adopt with confidence, and which field personnel could implement consistently and effectively.

3.2.5 Operational Experience Involving Losses of Electrical Inverters

AEOD case study C605 was initiated primarily because of the observed lack of overall improvement in the operational performance of safety- and nonsafety-related electrical inverters. Operating experience has shown that the loss of

an inverter is a frequent occurrence, with a total of 152 events involving inverter losses reported between 1981 and 1985.

Occurrences of events involving inverter losses raises two principal concerns: (1) increased frequency of plant transients, and (2) inoperable or improper functioning of both safety-related and other important plant equipment. Review and analysis of related operational data indicate that inverter loss events result from component failures and/or personnel actions. Further review indicates that a major contributing factor and/or cause for the occurrence of component failure events is an incompatibility between the actual inplant service conditions for inverters and their design service conditions. Approximately 50% of the events attributed to component failures were due in part or whole to high temperature and/or humidity within inverter enclosures and/or voltage spikes and perturbations at inputs and outputs of inverters. Contributing factors for events involving personnel actions included inadequate maintenance and testing procedures and deficient practices. In addition, inadequate planning, training, and verification for related maintenance and testing activities were contributing factors.

Overall, no significant decrease in the frequency of inverter losses has been detected on an industry-wide basis. Although the plants involved in these events have recovered without any serious consequences occurring, the frequency and potential consequences of these events continue to be of concern.

3.2.6 Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems

In reviewing operating experience from 1982 to 1986, AEOD case study C604 identified four events at four separate nuclear plants involving failures of solid state electronic components in safety-related instrumentation and control (I&C) systems due specifically to overheating.

Overheating of electronic components in safety-related I&C systems raises two concerns: (1) decreased reliability of electronic equipment due to increased failure rate of printed circuit cards and other heat sensitive electronic components, and (2) the potential for common cause failure of redundant safety-related instrument channels due to extended loss of normal cooling air flow to the cabinets in which the instruments are located. These concerns are generic to all operating nuclear plants that use solid state electronic components. Failures of instrument system components due to overheating have caused malfunctions of control systems, inoperability, and spurious actuation of protection and engineered safety features actuation systems (ESFAS) channels, inadvertent actuation and failure of an ESFAS train, and erroneous indications and alarms in the control room.

The review of the four events found that, in general, licensees did not readily identify elevated room ambient temperature or cabinet internal temperature as the root cause for the failure of the electronic components. The plant licensees, over an extended period, experienced several failures and many corrective actions before finally identifying overheating of components as the underlying reason for many of the failures they had experienced. A limited review of operating experience at other plants that have experienced electronic component failures of a similar type found that, in general, elevated room ambient or cabinet internal temperature was not identified as a possible root

cause of the failures. The study also found that plant technical specifications regarding area ventilation cooling systems and instrumentation systems have not adequately considered actual temperatures in the instrument cabinets. In addition, a review of the staff's proposed resolution of USI A-44 regarding design adequacy and capability of instrumentation and control system equipment needed to function in environmental conditions associated with a station black-out found that plant specific evaluations are needed with regard to the actual temperature and condition of heat sensitive components inside the instrument cabinets.

3.2.7 Operational Experience Involving Turbine Overspeed Trips

AEOD case study C602 was performed in response to the action item 8(f) of the action directed by the EDO to respond to the NRC staff investigation of the June 9, 1985 event at Davis-Besse. In this event, after main feedwater was lost, both auxiliary feedwater (AFW) turbines of the plant coincidentally tripped on overspeed, and resulted in a total loss of feedwater flow to the steam generators. The study evaluated past operating experience involving overspeed trips of PWR AFW turbine-driven pumps. The overspeed trips of AFW turbines were found to be somewhat wide-spread and one of the major causes for loss of operability or availability of AFW systems. A total loss of the AFW system when needed will significantly increase the probability of core damage.

In addition, the study was extended to include a review of turbine overspeed trip operating experience on BWR high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) turbine-driven pumps. Almost all of the turbines which are used in AFW, RCIC, and HPCI systems at operating plants are manufactured by Terry Company and are equipped with Woodward governors. The service conditions for these three systems are quite similar and, as such, review of HPCI and RCIC events aided in identifying causes for overspeed trips on AFW turbines. A total of 128 events involving overspeed trips of steam turbines associated with AFW, HPCI, and RCIC systems were reviewed. These events occurred between January 1972 and September 1985.

Review of these events indicates that the dominant attributed causes of AFW turbine overspeed trips are speed control problems associated with governors, and trip and reset problems associated with trip valves and overspeed trip mechanisms. These problems are primarily the result of inadequate performance by plant personnel, inadequate procedures, and insufficient system design considerations. The governor speed control problems are: (1) slow response of the Woodward Model PG-PL governor, (2) entrapped oil in the speed setting cylinder of the Woodward Model PG-PL governor, (3) incorrect governor setting, and (4) water induction into the turbine. The trip and reset problems stem from the complexity of reset operations and a lack of trip position indication.

3.3 Abnormal Occurrences Involving U.S. Nuclear Power Plants

Each calendar quarter, AEOD prepares and coordinates a Report to Congress on Abnormal Occurrences (AOs). AOs may be individual incidents, recurring events, generic concerns, or a series of incidents which the Commission determines are significant from the standpoint of public health or safety. The criteria for selecting AOs have not changed since publication in February 1977. Thus, the number of AOs per year can be viewed as a rough, yet reasonably constant index

to the performance of the nuclear power industry with regard to potentially serious or significant occurrences.

The number of AOs reported for nuclear power plants in the Reports to Congress for each calendar year since 1977 is shown in Figure 2. The number of AOs for 1986 is a preliminary estimate, since the last two quarterly reports have not yet received final staff concurrence and/or Commission approval. The history of AOs on a per plant basis is shown in Figure 3. An increasing trend in the number of AOs is noted starting in 1982, with a slight drop noted for 1985. One reason for this increase is related to the increasing number of nuclear power plants. A review of Figure 3 shows that the number of AOs per plant has remained relatively level for the past 4 years.

A significant fraction of AOs has been associated with plants licensed less than 2 years. For example, of the 49 AOs reported from 1981 through 1986, about 29% (14 events) occurred at plants licensed less than 2 years at the time of the AO.

A summary of 1986 abnormal occurrences, including those still under consideration, is provided in Appendix A. This summary includes power reactor, nonreactor, and medical misadministration abnormal occurrences. The two latter categories are discussed in Sections 4.1.1 and 4.2.4.

On November 18, 1986, AEOD coordinated an AO Reporting Meeting at headquarters (Bethesda), attended by representatives of most Headquarters and Regional Offices. These representatives were those generally involved in the AO reporting program (e.g., the Office AO Coordinators). The purposes of the meeting were to (1) assure consistency of reporting, (2) improve timeliness and quality of reporting, (3) reduce personnel resources, and (4) resolve questions concerning the program. The need for such a meeting was indicated by (1) the large turnover in AO Coordinators during the life of the AO program so that some coordinators may not be completely familiar with the policies and procedures, (2) staff questions regarding the threshold for reporting certain events, (3) uneven quality (content) of event writeups submitted to AEOD, and (4) the generally extensive time expended in identifying AOs and processing them.

The meeting was coordinated by members of AEOD who have been with the AO program since its inception in 1975; these individuals were the principal NRC staff coordinators for the development of the AO criteria, NRC AO Policy Statement, and the staff guidelines and procedure for the AO reporting process. From the questions and comments received, it was evident that the meeting was of mutual benefit to all of the participants.

Conclusions

In summary, the number of significant items of this magnitude remains relatively constant.

Figure 2
U. S. NUCLEAR POWER PLANTS
ABNORMAL OCCURRENCES VS. YEAR

Figure 3
U. S. NUCLEAR POWER PLANTS
ABNORMAL OCCURRENCES/PLANT VS. YEAR

3.4 Analysis of the 1986 Licensee Event Reporting

On January 1, 1984, the LER rule (10 CFR 50.73) became effective. As a result, the NRC and the industry have a description by the licensee, in a reasonably complete and detailed manner, of: all actuations of Engineered Safety Features (ESFs), including scrams; all losses of safety function at a system level; all significant systems interactions; all technical specification shutdowns/violations; and all significant internal and external threats to plant safety. The number of LERs submitted by year is shown in Table 3.

Table 3
LERs Submitted By Year

Year	LERs	Units	LERs Per Unit
1981	4016	75	53
1982	4400	81	54
1983	4839	84	57
1984	2435	92 (1)	26
1985	2997	97 (2)	31
1986	2818	104 (3)	27

- (1) Palo Verde not included - licensed 12/31/84.
- (2) Dresden 1, Humboldt Bay, and Three Mile Island 2 are not included in the 1985 data.
- (3) Dresden 1, Humboldt Bay, Three Mile Island 2, Seabrook 1 and Braidwood 1 are not included in the 1986 data.

The following discussion provides an overview of the LERs submitted in 1986. As noted on a number of occasions, we attach no safety significance to the raw number of LERs per se. The safety significance is assessed through engineering review of each LER, and through trends and patterns analysis of the content of the LERs. Variations in LER counts from plant to plant can result from a host of factors, only one of which is an actual difference in safety performance. Thus, the reporting pattern is examined with attention to both high and low reporters, in order to gauge the reporting process itself.

Analysis of the 2818 LERs submitted in 1986 indicates a broad range in the number of LERs submitted by each licensee (from three LERs to 93 LERs in this 1-year period). The mean number of LERs per unit is 27, while the median is 24. The distribution of LERs per unit is shown in Figure 4.

In terms of what was reported, the largest number of reports (44%) was associated with scrams and ESF actuations. The number is roughly split between ESF actuations (other than scrams), and scrams. A scram was reported in about 20% of the 1986 LERs, compared to only 1% of the 1981-1983 LERs (note: scrams were not specifically reportable prior to 1984).

The second most frequently reported type of event (31%) was a condition prohibited by technical specifications (TSs) or a shutdown required by the TS. This category covers conditions ranging from a missed surveillance test to the completion of a plant shutdown because of the unavailability of required safety components.

Figure 4

The third most frequent LER category concerned events that did or could have resulted in a loss of a safety function at the system level. This condition was reported in 7% of the LERs. The percent of LERs by the individual LER report requirement is shown in Table 4.

Table 4
LER Reporting by Report Requirement

Reference	Requirement	Percent*
50.73(a)(2)(iv)	RPS/ESF Actuation	44
50.73(a)(2)(i)	TS Shutdown or TS Violation	31
50.73(a)(2)(v)	Real or Potential Loss of a Safety System	7
50.73(a)(2)(ii)	Unanalyzed Conditions	4
50.73(a)(2)(vii)	Failures in Multiple Systems	3
50.73(a)(2)(iii)	External Threat	1
Other associated reporting requirements (e.g., Part 21, 50.36, 73.71, voluntary)		11

*Percents sum to greater than 100% because an LER can be reported under more than one reporting requirement.

When the LERs are assessed in terms of other classifications, such as NSSS vendor or architect-engineer, typically a wide variation results. Specific details are provided in Table 5.

Table 5
Average Number of LERs by Major NSSS Vendors in 1986

NSSS Vendor	Number of Units	Average Number of LERs Per Plant
B&W	8	18
CE	14	22
W	43	25
GE	37	34

The analysis of 1986 LERs is continuing. A number of major studies are in progress or planned in order to better characterize and understand the nature and significance of these operational events. The preliminary results of some of these studies (e.g., scrams and ESF actuations) are reported in Sections 3.6 and 3.7.

3.5 Assessment of Licensee Event Report Quality

The Licensee Event Report (LER) is one of the most important and widely analyzed documents in the nuclear industry. It is the principal means of identifying and analyzing safety problems and concerns which may not be recognized or properly understood as to potential significance. In order to

evaluate the overall quality of the contents of the LERs, a representative sample of a licensee's LERs is evaluated against the reporting requirements contained in 10 CFR 50.73. Each sample consists of:

- ° LERS with event dates during the assessment period.
- ° From the LERs submitted during the period, a maximum of 15 (selected randomly) are evaluated. If the licensee has submitted fewer than 15 LERs for the period, all of the LERs for the period are assessed.

Beginning in early 1986, for a multi-unit site where the LERs for the site are prepared by a single organizational entity, a single sample that meets the criteria described above is generally used for the entire site evaluation. Prior to 1986, a separate report was prepared for each unit at a multi-unit site.

The assessment of LER quality is performed by an AEOD contractor, Idaho National Engineering Laboratory (INEL) at Idaho Falls, Idaho. The evaluations performed by the contractor consist of a detailed review of each selected LER to evaluate the content of its text, abstract, and coded fields. The evaluation process for each LER is divided into two parts. The first part of the evaluation consists of commenting on the content of each LER. The second part consists of determining a score (0-10 points) for the text, abstract, and coded fields of each LER.

The LER-specific comments serve to point out what the analysts considered to be the specific deficiencies or observations concerning the information pertaining to the event, and also provide a basis for evaluating the general deficiencies for the overall sample of LERs that was reviewed.

A separate report is prepared for each licensee which includes a detailed discussion of the strengths and weaknesses of each LER. AEOD asks each Region to forward these reports to the licensee to assist in the preparation of future LERs. In addition, a brief summary statement of the results of the assessment is presented for each licensee. Graphic presentations of the distribution of the overall grades for the licensees assessed as of the end of December for 1985 and 1986 are shown in Figures 5 and 6, respectively.

It is noted that there has been a favorable shift in the number of units/stations receiving higher overall average scores during 1986. A significant portion of this increase is attributed to licensees which have been evaluated more than once. For example, the plant scoring 6.6 in 1985 scored above the industry average when reevaluated during 1986. (The plant scoring 6.6 in 1986 had not been evaluated previously.) Almost without exception during 1986, the overall average score showed improvement for each licensee which had been previously evaluated. Nevertheless, there is still considerable room for improvement for the majority of licensees to provide LER reports of consistently high quality. Future consideration will be given to varying the frequency of review based on the results of the evaluations.

During the December 1986 Region II Resident Inspectors Meeting, a presentation was given of the methods used to evaluate the quality of LERs, and of the results obtained for each licensee. Similar presentations are planned for other Regions.

Figure 5. **Distribution of overall average LER scores
as of December 1985**

Figure 6. **Distribution of overall average LER scores
as of December 1986**

Conclusion

There was general improvement in the quality of LERs during the year. However, there still is a rather wide divergence in the quality of LERs submitted by the licensees. In addition, there is still considerable room for improvement for many licensees.

3.6 Reactor Scram Experience In 1984, 1985, and 1986

This section analyzes unplanned reactor trips (i.e., scrams) which occurred at U.S. light water power reactors in 1984, 1985, and 1986. Data on reactor scrams were extracted from LERs submitted by licensees in conformance with 10 CFR 50.73. The results from completed studies of 1984 and 1985 scram data (AEOD/P504 and AEOD/P602, respectively) and preliminary results for a similar study for 1986 experience, are summarized below.

There are generally three phases to a scenario or sequence of events involving a reactor scram. First, there is the generation of some off-normal plant state which results in operation of the reactor protection system (RPS) or the need for a manual scram. Second, there is the operation of the RPS and control rod drive system. Third, there is the plant and operator response to the scram. Each phase has safety significance. For example, the NRC has concluded that a reduction in the frequency of challenges to plant safety systems should be a prime goal of each licensee, and that large Anticipated Transient Without Scram risk reductions can be achieved by reducing the frequency of transients which call for the RPS to operate.

3.6.1 1986 Scram Experience

A reactor scram is defined as an actuation of the RPS, whether automatic or manual, that results in control rod motion. Plants were included in these statistics if they: (1) held a full power operating license, and (2) accumulated critical hours for some portion of the calendar year in question. Reactor years were calculated for portions of the calendar year where necessary, based on the date of initial criticality.

Based on preliminary data assessment, in 1986 there was a total of 469 unplanned scrams at 93 U.S. LWRs which were licensed to operate at above 5% power and which had accumulated some critical hours. The corresponding figures for 1985 were 552 scrams at 92 LWRs.

Of the 469 scrams in 1986, a total of 52 (11%) were manual. The number of total scrams is down in 1986 as compared to 1985; however, the industry scram rate, while reduced, remained at nearly one scram per thousand hours of critical operation.

In 1986, the industry average rate (number per plant per year) for automatic trips showed its first significant reduction. The comparative statistics for 1983 through 1986 are as follows:

<u>Scram Type</u>	<u>Average Number Per Plant Per Year</u>			
	<u>1983*</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>
Manual	0.9	0.7	0.6	0.6
Automatic	5.6	5.2	5.4	4.5
Total	6.5	5.9	6.0	5.1

*From the NRC (NRR) study, "1983 Reactor Trip Statistics," dated August 1984.

3.6.2 Reactor Scram Frequency

Reactor scram rates for unplanned reactor scrams occurring in 1984, 1985, and 1986 are displayed for the major NSSS vendors in Table 6. Overall, there was a significant reduction in the overall PWR aggregate rate. There was a decrease in the total number of reactor scrams from 1985 to 1986 (reduction of 83 scrams), while the total critical hours and number of plants remained almost constant.

Table 6
Average Reactor Scram Frequency Data

	<u>1984</u>		<u>1985</u>		<u>1986</u>	
	<u>Scrams Per Plant Per Reactor Year</u>	<u>Scrams Per Plant Per 1000 Critical Hours</u>	<u>Scrams Per Plant Per Reactor Year</u>	<u>Scrams Per Plant Per 1000 Critical Hours</u>	<u>Scrams Per Plant Per Reactor Year</u>	<u>Scrams Per Plant Per 1000 Critical Hours</u>
PWR						
W	7.1	1.22	6.8	1.04	5.5	0.84
CE	5.9	0.86	7.5	1.24	6.2	0.91
B/W	3.0	0.44	5.0	0.88	2.4	0.46
Total	6.3	1.04	6.7	1.06	5.3	0.82
BWR						
GE	5.5	1.12	5.3	0.94	4.4	0.80

The decrease in the Westinghouse (W) average scram rate from 1.04 to 0.84 scrams per 1000 critical hours is reflective of a general decrease of the individual plant scram rates for plants operating in both years, as well as relatively low scram rates for units which achieved initial criticality in 1986. The two Westinghouse units achieving initial criticality in 1986 had a trip rate below two trips per 1000 critical hours. Overall there were 37 Westinghouse-designed plants with data for 1985 and 1986; 19 showed a decrease in the number of scrams, 15 showed an increase, and three registered no change.

The major cause for the reduction at CE-designed plants was the major reduction in the number of scrams at Waterford 3, a net reduction of 21 scrams. Seven other CE units also experienced a reduction in the overall scram rate.

The Babcock & Wilcox (B&W) average is based on the smallest number of plants (eight in 1985 and seven in 1986), and hence is most sensitive to individual plant behavior. In 1986, four of the seven plants experienced a reduction in their overall trip rate.

Finally, the General Electric (GE) BWR average shows a significant decrease from 0.94 to 0.80 scrams per 1000 critical hours. The decrease reflects a drop in the scram rate at 15 of the 28 GE-designed plants that operated in both 1985 and 1986.

In 1984, 1985, and 1986, the majority of reactor scrams occurred with the reactor power above 15% (i.e., 68% in 1984, 74% in 1985, 76% in 1986). The 1986 data is consistent with the 1985 results, which we believe reflect the short time spent at lower power. Over the past 3 years, 31%, 38%, and 38% of the total scrams in 1984, 1985, and 1986, respectively, occurred while the plant was at 95% power or above. Because the overwhelming majority of scrams occur at power levels greater than 15%, and because of the inherent greater decay heat removal requirements at higher power levels, our analysis focused on the power level greater than 15% power.

In 1984, a scram frequency of 2.0 scrams (above 15% power per 1000 critical hours) was selected as a breakpoint for examining relatively high scram rates. Figure 7 provides a plot of each plant's reactor scram rate for scrams above 15% power. The axes are (X) scrams per 1000 critical hours and (Y) the number of critical hours. In addition, horizontal and vertical reference lines bound plants into quadrants. The horizontal reference line represents a plant availability of approximately 70%. Plants above the 70% availability line and to the left of the vertical two scrams per 1000 critical hours reference line have high availability and low trip rates. Plants bound in other quadrants represent those plants with high availability/high scram rate, low availability/low scram rate, and low availability/high scram rate. Table 7 lists plants with high availability and low scram rates. Table 8 is provided to list those plants that are currently licensed but had no critical hours in 1986. In 1986, there was a modest increase in the average number of critical hours; i.e., 5878 critical hours in 1985 to 6016 critical hours in 1986.

Table 9 is provided to show plant by plant scram rates for 1986. The criteria for relatively high trip experience was met in 1984 by ten plants, in 1985 by eight, and in 1986 four plants met this criteria. The maximum rate in 1986 was 3.0 scrams per 1000 critical hours versus the 5.7 scrams per 1000 critical hours in 1984 and 4.8 in 1985. Only one of the four plants that exceeded the cutoff achieved initial criticality in 1986 (i.e., achieved criticality 4/18/86).

In conclusion, the scram frequency is noticeably improved (lower) for 1986 as compared to 1985. The number of plants exhibiting relatively high scram rates (based on their scram rate above 15% power exceeding 2.0 scrams per 1000 critical hours) has decreased significantly since 1984.

3.6.3 Initiating Systems (Above 15% Power)

As in previous years, each scram has been examined to determine the system containing the root cause of the scram. That system is defined as the "initiating system." The scram that occurred may have been the result of

Figure 7

1986 REACTOR SCRAM RATES VS CRITICAL HOURS BY PLANT

POWER GREATER THAN 15%

1985 REACTOR SCRAM RATES VS CRITICAL HOURS BY PLANT

POWER GREATER THAN 15%

Table 7

PLANTS WITH HIGH CRITICAL HOURS
RATE LESS THAN 2 TRIPS PER 1000 HRS
YEAR 1986

NAME	CRITHRS	RATE
ARKANSAS 2	6370.0	0.63
BEAVER VALLEY 1	6243.8	0.32
BIG ROCK POINT	8387.3	0.12
BRUNSWICK 1	8317.7	0.60
BYRON 1	7820.9	0.51
CALLAWAY 1	7307.6	0.82
CALVERT CLIFFS 1	6906.2	0.58
CALVERT CLIFFS 2	8442.0	0.59
COOK 1	7536.4	0.53
COOPER	6570.1	0.30
DIABLO CANYON 2	7296.3	1.37
DRESDEN 2	7110.1	0.42
DUANE ARNOLD	7348.2	0.00
FARLEY 1	7276.4	0.55
FARLEY 2	7549.7	0.40
FITZPATRICK	8075.8	0.25
FT. CALHOUN 1	8485.2	0.24
GINNA	7716.3	0.52
HATCH 2	6451.9	0.77
INDIAN POINT 3	6581.6	1.22
KEWAUNEE	7584.3	0.26
LASALLE 2	6614.0	0.45
LIMERICK 1	7146.1	0.14
MAINE YANKEE	7791.1	0.77
MILLSTONE 1	8276.5	0.24
MILLSTONE 2	6599.6	0.61
MILLSTONE 3	6201.1	0.81
MONTICELLO	6984.9	0.29
NORTH ANNA 1	7560.0	0.66
NORTH ANNA 2	7301.3	0.41
OCONEE 2	7253.7	0.69
OCONEE 3	7835.4	0.26
PEACH BOTTOM 2	7272.8	0.69
POINT BEACH 1	7905.4	0.25
POINT BEACH 2	7262.7	0.14
PRAIRIE ISLAND 1	7898.4	0.13
PRAIRIE ISLAND 2	7972.1	0.38
ROBINSON 2	7118.9	0.98
SALEM 1	7097.2	1.13
SAN ONOFRE 2	6480.0	1.08
SAN ONOFRE 3	7422.3	0.54
ST. LUCIE 1	8424.0	0.36
ST. LUCIE 2	7326.7	0.41
SUMMER 1	8453.2	0.59
SURRY 1	6233.2	0.64
TMI-1	6268.6	0.48
TROJAN	7064.1	0.57
TURKEY POINT 3	6988.4	0.86
WATERFORD 3	7011.6	0.86
WNP-2	6391.5	0.78
WOLF CREEK 1	6523.7	0.77
YANKEE ROWE	8343.5	0.36
ZION 2	7783.5	0.51

Table 8
 LICENSED REACTORS WITH ZERO CRITICAL HOURS
 YEAR 1984

----- VENDOR=BABCOCK WILCOX -----	
OBS	NAME
1	RANCHO SECO
2	THI-2*
----- VENDOR=WESTINGHOUSE -----	
OBS	NAME
3	SEQUOYAH 1
4	SEQUOYAH 2
5	HARRIS 1
6	BYRON 2
7	BRAIDWOOD 1
8	SEABROOK
----- VENDOR=GENERAL ELECTRIC -----	
OBS	NAME
9	DRESDEN 1*
10	HUMBOLDT BAY*
11	BROWNS FERRY 1
12	BROWNS FERRY 2
13	BROWNS FERRY 3
14	NINE MILE POINT 2
15	CLINTON 1

*Plants are permanently shut down.

Table 9
REACTOR TRIP RATES
YEAR 1986

NAME	MANUAL	AUTO MATIC	LESS THAN OR EQUAL 15% POWER	GREATER THAN 15% POWER	CRITICAL HOURS	TRIP RATE PER 1000 HOURS POWER GT 15	MEAN TIME BETWEEN TRIPS POWER ST 15%
ARKANSAS 1	0	4	2	2	5536.7	0.36	2768.4
ARKANSAS 2	0	5	1	4	6370.0	0.63	1592.5
BEAVER VALLEY 1	1	3	2	2	6243.8	0.32	3121.9
BIG ROCK POINT	0	1	0	1	8387.3	0.12	8387.3
BRUNSWICK 1	1	6	2	5	8317.7	0.60	1663.5
BRUNSWICK 2	0	2	0	2	4232.5	0.47	2116.3
BYRON 1	2	4	2	4	7820.9	0.51	1955.2
CALLAWAY 1	1	6	1	6	7307.6	0.82	1217.9
CALVERT CLIFFS 1	0	4	0	4	6906.2	0.58	1726.6
CALVERT CLIFFS 2	3	3	1	5	8442.0	0.59	1688.4
CATAWBA 1	0	9	2	7	5425.2	1.29	775.0
CATAWBA 2	2	5	2	5	2678.4	1.87	535.7
COOK 1	0	5	1	4	7536.4	0.53	1884.1
COOK 2	0	3	1	2	5560.5	0.36	2780.3
COOPER	0	2	0	2	6570.1	0.30	3285.0
CRYSTAL RIVER 3	0	1	0	1	3691.1	0.27	3691.1
DAVIS-BESSE 1	0	1	1	0	178.0	0.00	.
DIABLO CANYON 1	0	3	1	2	5967.4	0.34	2983.7
DIABLO CANYON 2	1	10	1	10	7296.3	1.37	729.6
DRESDEN 2	2	4	3	3	7110.1	0.42	2370.0
DRESDEN 3	1	5	0	6	2766.4	2.17	461.1
DUANE ARNOLD	1	1	2	0	7348.2	0.00	.
FARLEY 1	0	4	0	4	7276.4	0.35	1819.1
FARLEY 2	0	4	1	3	7549.7	0.40	2516.6
FERMI 2	2	2	3	1	1403.0	0.71	1403.0
FITZPATRICK	0	2	0	2	8075.8	0.25	4037.9
FT. CALHOUN 1	1	1	0	2	8485.2	0.24	4242.6
GINNA	1	3	0	4	7716.3	0.52	1929.1
GRAND GULF 1	1	5	2	4	5624.6	0.71	1406.1
HADDAM NECK	3	3	2	4	5060.9	0.79	1265.2
HATCH 1	0	4	0	4	5521.2	0.72	1380.3
HATCH 2	0	7	2	5	6451.9	0.77	1290.4
HOPE CREEK 1	2	9	8	3	2669.8	1.12	889.9
INDIAN POINT 2	2	8	0	10	5101.9	1.96	510.2
INDIAN POINT 3	0	9	1	8	6581.6	1.22	822.7
KEMAUNEE	0	3	1	2	7584.3	0.26	3792.1
LACROSSE	2	16	10	8	4400.8	1.82	550.1
LASALLE 1	0	2	1	1	2395.7	0.42	2395.7
LASALLE 2	0	5	2	3	6614.0	0.45	2204.7
LIMERICK 1	0	1	0	1	7146.1	0.14	7146.1
McGUIRE 1	0	3	0	3	5022.2	0.60	1674.1
McGUIRE 2	1	5	1	5	5770.4	0.87	1154.1
MAINE YANKEE	2	4	0	6	7791.1	0.77	1298.5
MILLSTONE 1	1	2	1	2	8276.5	0.24	4138.3
MILLSTONE 2	0	4	0	4	6599.6	0.61	1649.9
MILLSTONE 3	0	14	9	5	6201.1	0.81	1240.2
MONTICELLO	0	2	0	2	6984.9	0.29	3492.4
NINE MILE POINT 1	0	2	2	0	5823.5	0.00	.
NORTH ANNA 1	2	5	2	5	7560.0	0.66	1512.0
NORTH ANNA 2	0	4	1	3	7301.3	0.41	2433.8
OCONEE 1	0	2	0	2	5948.7	0.34	2974.4
OCONEE 2	0	5	0	5	7253.7	0.69	1450.7

Table 9 (cont'd)
REACTOR TRIP RATES
YEAR 1986

NAME	MANUAL	AUTO MATIC	LESS THAN OR EQUAL 15% POWER	GREATER THAN 15% POWER	CRITICAL HOURS	TRIP RATE PER 1000 HOURS POWER GT 15	MEAN TIME BETWEEN TRIPS POWER GT 15%
OCONEE 3	1	1	0	2	7835.4	0.26	3917.7
OYSTER CREEK	0	3	2	1	2389.1	0.42	2389.1
PALISADES	0	2	0	2	1490.5	1.34	745.3
PALO VERDE 1	0	13	0	13	5498.0	2.36	422.9
PALO VERDE 2	0	10	2	8	3803.1	2.10	475.4
PEACH BOTTOM 2	1	4	0	5	7272.8	0.69	1454.6
PEACH BOTTOM 3	3	6	3	6	5919.3	1.01	986.5
PERRY 1	1	4	5	0	1429.6	0.00	.
PILGRIM 1	0	4	3	1	1715.5	0.58	1715.5
POINT BEACH 1	0	2	0	2	7905.4	0.25	3952.7
POINT BEACH 2	0	2	1	1	7262.7	0.14	7262.7
PRAIRIE ISLAND 1	0	2	1	1	7898.4	0.13	7898.4
PRAIRIE ISLAND 2	0	3	0	3	7972.1	0.38	2657.4
QUAD CITIES 1	0	4	2	2	6151.3	0.33	3075.6
QUAD CITIES 2	0	2	0	2	5728.0	0.35	2864.0
RIVER BEND 1	0	18	2	16	5305.7	3.02	331.6
ROBINSON 2	0	11	4	7	7118.9	0.98	1017.0
SALEM 1	0	9	1	8	7097.2	1.13	887.2
SALEM 2	0	9	3	6	5629.3	1.07	938.2
SAN ONOFRE 1	0	2	1	1	2975.1	0.34	2975.1
SAN ONOFRE 2	1	7	1	7	6480.0	1.08	925.7
SAN ONOFRE 3	1	5	2	4	7422.3	0.54	1855.6
SHOREHAM 1	0	0	0	0	1299.5	0.00	.
ST. LUCIE 1	1	3	1	3	8424.0	0.36	2808.0
ST. LUCIE 2	1	4	2	3	7326.7	0.41	2442.2
SUMNER 1	0	6	1	5	8453.2	0.59	1690.6
SURRY 1	3	2	1	4	6233.2	0.64	1558.3
SURRY 2	1	3	0	4	6171.1	0.65	1542.8
SUSQUEHANNA 1	0	0	0	0	6196.3	0.00	.
SUSQUEHANNA 2	1	1	1	1	5946.6	0.17	5946.6
TMI-1	0	4	1	3	6268.6	0.48	2089.5
TROJAN	0	4	0	4	7064.1	0.57	1766.0
TURKEY POINT 3	1	5	0	6	6988.4	0.86	1164.7
TURKEY POINT 4	1	2	1	2	3048.1	0.66	1524.1
VERMONT YANKEE	0	2	2	0	4359.6	0.00	.
WATERFORD 3	0	7	1	6	7011.6	0.86	1168.6
WNP-2	0	7	2	5	6391.5	0.78	1278.3
WOLF CREEK 1	0	7	2	5	6523.7	0.77	1304.7
YANKEE ROWE	0	4	1	3	8343.5	0.36	2781.2
ZION 1	0	2	0	2	5491.0	0.36	2745.5
ZION 2	0	4	0	4	7783.5	0.51	1945.9
	==	===	===	===			
	52	417	123	346			

hardware failure; inappropriate system operation, maintenance, or calibration; or testing of the initiating system. The results of these categorizations are shown in Figure 8. The data for the major balance-of-plant (BOP) systems (i.e., feedwater, turbine, condensate, main generator, and main steam) are provided in Figure 9. Fifty-seven percent of all reactor scrams at power levels greater than 15% were due to BOP systems in 1986. Nearly half of the BOP trips were attributed to the main feedwater system. The scram frequency (in terms of scrams per thousand critical hours) attributed to various initiating systems for 1986 is consistent with and follows the same general trend seen in 1984 and 1985. AEOD's ongoing analysis of the 1986 scram data will provide additional details about root causes, component failures, and personnel interactions as they relate to initiating systems.

3.6.4 Causes of Scrams (Above 15% Power)

The LER description of each scram was reviewed to determine the general classification of cause or causes, and the results are shown in Figure 10. As has been the trend in past years, hardware failures dominated in 1986. Personnel related problems (i.e., human error, manual steam generator control problems and procedural deficiencies) in 1986 accounted for 29% of all reactor scrams above 15% power, which is similar to the experience in 1984 and 1985.

Unlicensed personnel (technicians, contractors, nonlicensed operating staff, and other utility staff personnel) were responsible for 16% of all scrams above 15% power; this represents a small increase from the 14% found in 1985. In 1985 technicians were involved in approximately one of every 12 scrams; in 1986, one in every nine scrams.

The four activities (maintenance, testing, calibration and troubleshooting) require interaction of personnel with plant systems. During 1986, maintenance, testing, and calibration were responsible for 36% of all scrams that occurred. This compares with 30% of all scrams when these activities were ongoing in 1984 and 1985. In 1986, when troubleshooting is added, the figure is raised by 4%. It is noteworthy that in 1986, for all scrams above 15% power attributed to human error, 79% were caused by human errors during maintenance, testing, calibration, and troubleshooting activities. Unlicensed personnel (technicians) were responsible for 63% of the human errors during these activities.

Hardware failures were encountered in 25% of the cases during these evolutions. Figure 11 is provided to display the systems most frequently impacted during maintenance, testing, troubleshooting and calibrations. Testing only was responsible for 22% of all scrams in 1986. Testing associated with the RPS and turbine were responsible for the majority of these scrams (12% of all scrams). A detailed analysis of these scrams will be provided in the AEOD 1986 scram report.

3.6.5 Scrams With Associated Failures

Unplanned scrams where the recovery was complicated by additional equipment failures or personnel error can be of concern because of the higher level of stress and demands placed upon the operating personnel and mitigating systems. We define "associated failures" as component failure or personnel error that did not contribute directly to the cause of the scram, but are associated with post scram recovery (e.g., normally the failure was discovered or occurred when the component was actuated to mitigate the consequences of the scram).

INITIATING SYSTEM SUMMARY

POWER GREATER THAN 15%

Figure 8

PRIMARY BALANCE OF PLANT SYSTEMS POWER GREATER THAN 15%

Figure 9

CAUSE SUMMARY POWER GREATER THAN 15%

Figure 10

SYSTEMS IMPACTED MOST FREQUENTLY

BY
TESTING MAINTENANCE TROUBLESHOOTING CALIBRATIONS
POWER GREATER THAN 15%
YEAR 1986

OTHER INCLUDES REMAINING SYSTEMS INITIATING SCRAMS

Figure 11

In 1986 associated failures were experienced in 18% of scrams. The comparative statistics for the period 1984 through 1986 are shown below:

<u>Year</u>	<u>Scrams with Associated Failures</u>	<u>Percentage of all Scrams With Associated Failures</u>
1984	77	20
1985	109	23
1986	66	18

3.6.6 Quantitative Safety Significance Measures

In 1985, a concurrent program was in place to quantitatively measure the safety significance of an event sequence (transient) involving a reactor scram. The specialized probabilistic event tree technique called "Accident Sequence Precursor (ASP)" was used to evaluate the 1985 reactor scram sequences as documented in LERs. (See Section 5.2.6 for a description of the ASP program.) Our use of this technique was exploratory in 1985, and results are not yet available for scrams occurring in 1986.

3.7 Engineered Safety Features (ESF) Actuations

Safety systems should work reliably and properly when challenged, and should not be challenged frequently or unnecessarily. In order to gain an understanding regarding the need and frequency of the challenges to safety systems, the Commission required, as part of the revised LER system, that actuation of an engineered safety feature (ESF) be reported to the NRC as an LER. This reporting requirement became effective on January 1, 1984. Prior to this date, the actuations of these systems were not directly reportable.

ESF systems are designed to control and mitigate specific occurrences that might challenge the integrity of the reactor and/or adversely affect plant personnel or the general populace. Generally, these include systems designed to control reactor core reactivity, isolate and cool containment, supply emergency cooling to the reactor fuel, remove residual core heat, assure habitability of the control room under all conditions, control radioactivity releases to the environment, and provide a source of emergency power.

As part of the AEOD trends and patterns analysis program, ESF actuations are the subject of periodic detailed review. A study of ESF actuations for the first half of 1984 was published as AEOD/P503 in August 1985. The analysis of the second half of 1984 was completed and was issued in August 1986 as AEOD/P603. Data for 1985 were compiled, and analysis results will be issued in 1987. A summary of the findings for 1985, and a discussion of the trends from 1985 to 1986 (based on preliminary data), are presented below.

3.7.1 Frequency of ESF Actuations

It was found that 1582 ESF actuations were reported in 1985 involving 91 of the 97 reactors that were eligible for reporting per 10 CFR 50.73. Of the 97 units eligible, six units (6%) experienced no actuations during the year, 31 units (32%) had between one and five actuations, 18 units (19%) had between six and

ten actuations, eight units (8%) had between 11 and 15 actuations, six units (6%) had between 16 and 20 actuations, and 28 units (29%) experienced more than 20 ESF actuations, with the maximum number of actuations at any one unit being 86. Figure 12 shows the unit distribution of total ESF actuations for 1984, 1985, and 1986. Only about 9% of all reported ESF actuations involved an emergency core cooling system (ECCS), and none of these occurrences were necessary to control an actual loss-of-coolant accident (LOCA). Over 80% of the actuations that occurred in ESF systems were associated with either an isolation function or a ventilation function.

3.7.2 Valid ESF Actuations

In 407 of the 1582 cases studied (26%), the measured parameter reached the intended setpoint for ESF actuation; however, in only 15 of these 407 cases were the actuations considered needed in terms of providing a required ESF system response for protection from an actual design basis event. These 15 "design basis" ESF actuations represented less than 1% of the 1582 cases. They occurred at 12 different units, with no unit having more than three events. They included eight cases involving loss of offsite power resulting from a loss of a bus, four events involving a control building heating, ventilating, and air conditioning system, and five high radiation events.

In the remaining 392 valid cases, the ESF actuations resulted from conservative alarm setpoints, equipment failures, or personnel errors. These ESF actuations were considered to be valid (i.e., the measured parameter reached the intended actuation setpoint), but did not represent a response to a design basis event. Rather, they were actuations resulting from non-design basis conditions, such as radioactive trash being moved near a radiation monitor. These valid but non-design basis actuations were primarily associated with water level (reactor water level in BWRs and steam generator level in PWRs), loss of power, flow (about 90% were reactor water cleanup flow), or toxic gas monitors. Most of them were associated with isolation of the reactor water cleanup system or the containment, or the ESF mode of actuation of the control room ventilation. The distribution of the ESF system functions involved and the measured parameters (signals) for valid ESF actuations are shown in Table 10.

3.7.3 ECCS Actuations

In 144 of the ESF actuations at 62 units, an ECCS was actuated. Only 96 resulted in actual injection of fluid. These 96 actual injections occurred in 41 different units. Three units had more than four actual injections. The 96 actual injections were more than double the 43 actual injections in 1984. Of these 96 actual injection events, none were needed to control an actual LOCA.

3.7.4 False ESF Actuations

A total of 1175 false actuations, or 74% of the 1582 ESF actuations, occurred at 87 units. These false ESF actuations principally affected systems whose functions were associated with either isolation or ventilation. False ESF

UNIT DISTRIBUTION OF 1984/1985/1986 ESF ACTUATIONS

UNIT DISTRIBUTION OF 1984/1985/1986 ESF ACTUATIONS

Figure 12

UNIT DISTRIBUTION OF 1984/1985/1986 ESF ACTUATIONS

UNIT DISTRIBUTION OF 1984/1985/1986 ESF ACTUATIONS

Figure 12 (cont'd)

Table 10

Measured Parameters and Associated System
Functions for Valid ESF Actuations
Occurring in 1985

MEASURED PARAMETER (Signal)	General System Function					TOTAL
	FLUID	HEATING AND VENTILATION (HVAC)	ISOLATION	EMERGENCY POWER	NOT DETERMINED (MISC)	
MANUAL	7	1	3	1	1	13
LOW VOLT	21	18	18	36	1	94
LOW SG PRESSURE	1	0	3	0	0	4
LOW RX VESSEL LEVEL	15	4	26	2	0	47
LOW RX PRESSURE	4	1	2	0	0	7
HIGH STEAM LINE PRESSURE	1	0	1	0	0	2
HIGH RADIATION	8	53	28	0	1	90
FIRE	0	0	0	0	0	0
HIGH TOXIC GAS	0	3	0	0	0	3
HIGH RX WATER CLEANUP SYSTEM FLOW	0	0	22	0	0	22
HIGH ROOM DIFFERENTIAL TEMPERATURE	0	0	14	0	0	14
HIGH PRIMARY CONTAINMENT PRESSURE	0	1	0	0	1	2
LOW CONTAINMENT VACUUM	3	8	0	0	0	11
COMBINATION	65	20	48	20	0	153
UNKNOWN	0	0	0	0	1	1
HIGH STEAM FLOW	0	0	0	0	0	0
LOW SG LEVEL	24	1	1	0	1	27
HIGH RESIDUAL HEAT REMOVAL SYSTEM FLOW	0	0	1	0	0	1
RX TRIP	2	0	1	0	0	3
HIGH RX PRESSURE	5	0	2	0	2	9
HIGH SG LEVEL	5	0	6	0	0	11
TRIPPED MAIN FEED PUMP	16	0	6	0	2	24
OTHER	11	2	12	2	0	33
HIGH DRYWELL PRESSURE	0	0	0	0	0	0
ANY LOW LEVEL	1	0	0	0	0	1
LOW STEAM LINE PRESSURE	5	1	7	0	1	14
LOW CONDENSOR VACUUM	0	0	2	0	0	2
LOW SAMPLE FLOW	0	0	0	0	0	0
HIGH TEMPERATURE	1	0	2	0	0	3
TOTAL ACTUATIONS	196	113	205	61	23	597

actuations were caused mostly by spurious actuations, equipment failures, or problems related to personnel. The main parameters (signals) involved with these false actuations were radiation and loss of power.

3.7.5 ESF Failures

In only ten of the 1582 ESF actuations (less than 1%) did an ESF system (the entire system) fail to actuate properly. Only one unit had more than one such event.

In 78 additional cases the ESF actuations performed properly, but one or more failures were associated with the actuation. There was no identifiable trend or pattern for these failures, and redundant systems were available to perform required safety functions. Only three units had more than four such failures.

3.7.6 Conclusions Based Upon 1985 Data

Analyses of the 1985 data support the following general conclusions:

- ° Fifteen units were identified as experiencing repeated unresolved ESF actuations which could ultimately challenge continued equipment operability and proper personnel response.
- ° The remaining events necessitating ESF actuations, including ECCS actuations, have not been individually significant. This also holds true for failures and problems associated with the ESF actuations. In addition, it was readily apparent that the majority of the ESF actuations were unnecessary, and that the rate of these actuations could be greatly decreased by: (1) reducing the number of equipment failures during normal operation, (2) reducing the number of personnel errors during maintenance and testing, and (3) revising actuation setpoints to more appropriate protective levels. ESF functions associated with isolation or ventilation should receive first priority.

3.7.7 Use of ESF Actuations as Performance Indicators

The wide variety of ESF systems and the differences in the types of ESF actuations (including variations in immediate safety significance) make comparisons among units very difficult. As a result, only selected ESF actuations are being used as performance indicators [i.e., unplanned ECCS actuations that result from reaching an ECCS actuation setpoint or from a spurious or inadvertent ECCS signal, and the number of emergency ac power system actuations that result from the loss (deenergization) of a safeguard bus.]

3.7.8 Comparison of 1984, 1985, and 1986 Data

Data for 1986 is currently being analyzed in detail, and complete results are not available at this time. However, some preliminary trends can be described using the preliminary 1986 data. The average number of ESF actuations per reporting plant per unit of time increased from 1984 to 1985, but appears to be decreasing from 1985 to 1986.

The number of units reporting at least one ESF actuation increased from 1984 to 1985, and again from 1985 to 1986. The maximum number of actuations at a single unit decreased from 1984 to 1985, but increased from 1985 to 1986. Safety injections more than doubled from 1984 to 1985, but decreased dramatically in 1986. While valid actuations remained fairly level, false actuations increased greatly from 1984 to 1985, but decreased in 1986.

Assuming the 1986 preliminary data is confirmed, it appears that the rate of ESF actuations is decreasing for the first time since such events became reportable. Comparisons of ESF actuations for 1984, 1985, and 1986 are summarized in Table 11.

3.8 NPRDS Trends and Patterns Analysis Program

In 1986, AEOD began the full scale implementation of its NPRDS Trends and Patterns Analysis Program, a part of the AEOD Trends and Patterns Program. This program analyzes component failure data from the Nuclear Plant Reliability Data System (NPRDS), a data base voluntarily supported by the U.S. nuclear power plant (NPP) industry and maintained by the Institute of Nuclear Power Operations (INPO). The data base includes records that provide nameplate information on the design characteristics of the components as well as records describing their failures.

Major components within the main feedwater (MFW) system of pressurized water reactors (PWRs) were the first components selected for analysis. This system was selected because loss of MFW is a part of a majority of the dominant accident sequences in probabilistic risk assessments of PWRs. Furthermore, MFW transients are the major source of unplanned reactor scrams.

The goal of the trend and pattern analysis of MFW component failure data is to identify component attributes that are associated with relatively high incidences of failure. Manufacturer, valve type, and pump rotational speed are examples of component attributes under study; in addition, the pattern of failures among NPP units is studied.

The initial step in the trend and pattern analysis is a review of data with the utilities to verify or clarify any data that seem atypical or inconsistent. Although differences exist among PWR plant MFW systems, the common function of the specific components under study provides a relatively homogeneous population.

The second step is to select, from the attributes listed above, a set of component attributes (i.e., statistical variables) to analyze for their statistical relation to failure.

A series of statistical methods from the field of survival analysis is then applied to the data to identify trends and patterns in failures with regard to the variables. The methods use the component lifetimes (i.e., times between failures) and the component attributes. These methods identify possible

Table 11
ESF Actuations for 1984, 1985, and 1986

Category	1984	1985	1986*
Total ESF Actuations	1102	1582	1422
Units Eligible to Report	91	97	106
Units Reporting ESF Actuations	81	91	101
Units Reporting (%):			
<u>No. of ESF Actuations</u>			
0	11%	6%	5%
1-5	49%	32%	34%
6-10	16%	19%	16%
11-15	3%	8%	13%
16-20	3%	6%	8%
> 20	16%	29%	25%
Maximum ESF Actuations at Single Unit	152	86	104
ECCS Actuations (%)	8.4%	9.1%	6%
° No. of Actual Injections	43	96	55
° No. of Injections for LOCA	0	0	N/A
Isolation/Ventilation Actuations (%)	70%	80%	N/A
Valid ESF Actuations	400 (36%)	407 (26%)	418 (29%)
° Design Basis	34 (3%)	15 (1%)	33 (2%)
° Non-Design Basis	366 (33%)	392 (25%)	385 (27%)
False ESF Actuations	702 (64%)	1175 (74%)	1004 (71%)
* Preliminary Data			
N/A: Not available at this time.			

factors that influence the failure time distributions. For numerical attributes such as valve inlet size, a method tests for trends. All of these statistical methods are tailored to use truncated lifetimes (shortened by the data cutoff date or the study start date) as well as actual times between failures. In addition, the failures are studied in calendar time to detect shifts in the rate of component problems and identify specific attributes with rates significantly above an average baseline failure rate.

The final step is an engineering follow-up of the statistical results. The factors identified by the statistical analysis are starting points for this investigation. Failures for components with attributes flagged in the statistical analysis are reviewed, considering reported event details as well as component attributes. Root causes for these failures are sought through extensive discussions with appropriate plant personnel, NRC inspectors, and component manufacturers. By identifying the underlying causes and the practices that prevent recurrence, the benefits of experience can be shared to upgrade the MFW system performance for the entire PWR nuclear power plant industry.

To date, AEOD has studied MFW flow control valves, bypass flow control valves, and turbine-driven pumps. During 1986, three proprietary reports were prepared. The first two, "Failure Time Analysis of Feedwater Flow Control Valve and Valve Operator Components Within the Nuclear Plant Reliability Data System" and "Failure Time Analysis of Feedwater Flow Control Bypass Valve and Valve Operator Components Within the Nuclear Plant Reliability Data System" described the statistical analysis of these components. The third report, "Resolution of Feedwater Flow Control Valve and Feedwater Bypass Valve Issues Identified in NPRDS Trends and Patterns Analysis," discussed the follow-up engineering investigation and focused on resolving the issues identified in the statistical analysis and on identifying the practices and conditions that have been responsible for some of the problems. It gave recommendations for practices that help remedy these problems and minimize their recurrence.

The primary finding of the three studies issued in 1986 is that differences among units and stations have a greater influence on the performance of these components than any of the other component attributes studied. The reports on the flow control valves and the bypass flow control valves identified ten and six units, respectively, as outliers because of their frequent problems with these valves. All the units except one identified as having relatively high rates for bypass valve failures also were identified as having high reporting for full-power main feedwater flow control valve problems.

Results of the follow-up engineering investigation for the flow control valves and the bypass (startup) flow control valves are similar; therefore, they are discussed together. Table 12 presents the findings. Most of the recommendations involve improved maintenance and maintenance procedures, the upgrading of the instrument air system, the use of improved valve packing, and the use of components resistant to operating environments.

Table 12
Findings of the Flow Control Valve Studies

Problem	Cause	Actions to prevent problems
Valve operator failure	System or valve-induced vibration	Use flexible stainless steel instrument air lines. Use vibration-resistant connectors and fasteners (especially for the solenoid valves).
Valve operator	Oil, moisture and/or rust, or foreign particles in the instrument air system	Upgrade the instrument air system with improved blowdown valves and dryers. Monitor instrument air quality and establish maintenance schedules allowing prompt corrective action.
Valve operator failure	Outdoor weather conditions	Use waterproof solenoids.
Valve and valve operator failures	Poor maintenance procedures	Use detailed maintenance procedures that assure the completion of proper maintenance and adjustments before system startup. Provide adequate training and support of the maintenance personnel. Consult with valve manufacturers to establish efficient routine maintenance schedules. Have valve manufacturers refurbish the valve trim instead of doing this in-house. Cover disassembled valves during maintenance.
Valve released leakage	Packing leaks	Use new packing materials with low shrinkage and designs that maintain constant pressure on the packing (spring-loaded, for example).
	Bonnet/flange	In maintenance, carefully inspect the flange before reassembly.
Valve contained leakage	Improperly adjusted valve operators	Use improved, valve-specific maintenance procedures.
	Damaged valve trim (plug and cage or seats)	Use proper maintenance. Consult valve manufacturers for advice on improved valve trim designs and materials for actual plant conditions such as higher pressure drops.

In conclusion, the engineering evaluation of the statistical results from the trend and pattern analysis shows that proper maintenance and the use of appropriate materials dominate in avoiding problems with MFW components. Although the MFW system is not a safety system, upgrades of that system and supporting systems such as the control air and oil systems to make the MFW system more reliable will reduce reactor scrams and unnecessary demands on safety systems.

4. COMMENTS AND OBSERVATIONS ON 1986 OPERATING EXPERIENCE AT OTHER LICENSEES

During 1986, a number of events involving NRC and Agreement State licensed nonreactor facilities and activities were reported to the NRC. This section provides an overview and summary of reported events involving these nonreactor facilities, and medical misadministrations occurring in 1986.

4.1 Nonreactor Events

The AEOD Nonreactor Event Report (NRER) data base contains information on licensed nuclear materials and fuel cycle operational events and on personnel radiation exposure events. The NRER data base management system provides for input, storage, retrieval, and computer-assisted analyses of operational event data, and may be used to identify trends in operational safety events which may signal a need for remedial actions by the NRC and/or licensees.

4.1.1 Occurrences in 1986

The NRER data base includes 202 records of events that were entered into the data base during 1986. Information on these events was contained in reports submitted by nonreactor licensees to the Regional Offices or in other documents, primarily inspection reports. The data base does not include information from certain fuel cycle licensee reports, such as those related to routine effluent releases, nor does it include information from reports of medical misadministrations (a separate data base on this subject is maintained - see Section 4.2). Table 13 provides information on the types of licensees for which information was entered into the data base.

An NRER data base item may be associated with more than one category of event. For example, a report from a radiography licensee concerning a personnel radiation exposure would be counted in the total number of radiation exposure events as well as in the total number of events involving radiography. The 202 nonreactor licensee reports were cataloged as 317 entries in ten different areas (see Table 13 for details); in addition, reports received for which no program code was available, were delegated to the "Other" category. Note that, because some reports are associated with more than one event category, the total number of events exceeds the total number of reports.

Certain categories in Table 14 are primary categories; that is, they contain events from all types of licensees. These primary categories are: exposures; lost, abandoned, or stolen material; leaking sources; release of material; and consumer products. With few exceptions, most of these events are assigned to only one of the categories.

The secondary categories in Table 14 are designed to capture events by the type of licensee involved in the event. Many of the events assigned to these categories are also assigned to primary event categories. Secondary categories generally serve as a measure of the frequency with which certain types of licensees make reports to the NRC.

Table 13
Types of Licensees That Submitted Reports During 1986

<u>License Type</u>	<u>Number of Reports Received*</u>
Academic	2
Medical	41
Commercial/Industrial Measuring Systems	54
Well Logging (22)	
Other Measuring Systems (32)	
Manufacturing and Distribution (Excluding Medical)	12
Industrial Radiography	12
Single Location (In Plant) (3)	
Multiple Locations (Field) (9)	
Irradiator	3
R&D	12
Source Materials**	8
Mills (1)	
UF-6 Facilities (3)	
Other (4)	
Special Nuclear Material (Including Plutonium)	10
Agreement State	17
Other***	<u>31</u>
Total	202

*Medical misadministration reports are not included.

**Routine environmental effluent release reports, e.g., reports required by 40.65 and 70.59 were not included in the totals for source and special nuclear materials licensees.

***Number includes reports received for which no program code was available.

Table 14

Categorization of Nonreactor Event Reports Occurring During 1986

<u>Category*</u>	<u>Number of Reports Associated</u>
Primary Categories:	
Personnel Radiation Exposures	37
Lost, Abandoned, and Stolen Material	68
Leaking Sources	21
Release of Material	22
Consumer Products	4
Secondary Categories:	
Fuel Cycle (e.g., Mills, UF-6 Facilities Special Nuclear Material)	13
Industrial Radiography	19
Manufacturing and Distribution (Including Medical)	34
Commercial/Industrial Measuring Systems (Excluding Well Logging)	31
Other**	<u>68</u>
Total	317

*An NRER database item may be associated with more than one category of event, for example, a report from a radiography licensee concerning a personnel radiation exposure would be counted in the total number of radiation exposure events as well as in the total number of events involving radiography.

**Other includes categories such as medical, transportation, miscellaneous, etc.

4.1.2 Radiation Exposure Events

The NRER data base contains information from 37 reports of events that were entered during 1986 in which there was the potential for or an actual radiation overexposure. Of these 37 events, 13 involved actual radiation overexposures; a fourteenth event (Department of Air Force, Wright-Patterson Air Force Base) may have resulted in an overexposure. The types of licensees associated with the actual overexposures reported during 1986 were as follows:

<u>Licensee Type</u>	<u>Number of Overexposure Events</u>	<u>Total Number of Individuals Exposed</u>
Medical/Academic	2	2
Radiography	8	10
Commercial/Industrial	1	1
Fuel Cycle	1	2
Broad Scope	1	1 possible
General Licensee	<u>1</u>	<u>1</u>
Total	14	16 + 1 possible

- ° Medical/Academic Licensee - The two reports of overexposures at medical or academic licensees involved licensee personnel and included: two whole body exposures (3.4 rem/quarter and 21 rem). The reports came from different licensees.
- ° Radiography Licensee - All of the radiography overexposures were received by licensee personnel. Five of the eight reports were whole body exposures, and generally less than 5 rem/quarter. In a sixth event, occurring at a DOE site, film badges of two individuals showed whole body exposures of less than 5 rem. Extremity exposures for this event were unavailable at the time this report was prepared.

Two events occurred in prior years in Agreement States. In the first, an overexposure that was received in 1984 was not reported to the State until late in 1985. The radiographer may have received a hand exposure of 29,000 rem and a whole body exposure of 47 rem. In a second overexposure event, a radiographer received an estimated hand dose of 2000 rad and a whole body dose of about 6 rad. The overexposures occurred at different licensees.

- ° Commercial/Industrial Licensee - The commercial/industrial overexposure received by a licensee employee involved an extremity overexposure (42 rad).
- ° Fuel Cycle Licensee - The reported fuel cycle overexposure resulted in calculated internal exposures of two individuals that exceeded 40 MPC hours.

- ° Broad Scope Licensee - An individual involved in decontamination activities at Wright-Patterson Air Force Base inhaled radioactive material (Am-241) that may have resulted in an overexposure.
- ° General Licensee - A member of the public could have received a radiation exposure of from 0.6 to 1.7 rem (buttocks) and from 69-200 rem to the leg from a gauge. Effective control over the gauge had been lost.

In general, all of the 1986 overexposures were minor. There were two substantial overexposures to radiographers that occurred in prior years in Agreement States. These two events were reported as abnormal occurrences (see Appendix A for brief summaries of abnormal occurrences).

4.1.3 Lost, Abandoned, and Stolen Material

Licensees are required to report the loss or theft of licensed material that has occurred in such quantities and under such circumstances that it appears to the licensee that a substantial hazard may result to persons in unrestricted areas (10 CFR 20.402(a)(1)).

Sixty-eight events occurred during 1986 that involved lost, abandoned, or stolen licensed material. These events consist of 47 reports of lost or stolen material, plus 21 reports of abandoned, irretrievable well-logging sources. One of the 68 events resulted in a radiation overexposure. In this event, an Abnormal Occurrence, effective control over an industrial gauge was lost and a member of the public was overexposed.

Of the 47 reports of lost or stolen sources, 21 were found, two pacemakers were lost (one probably was buried and the other was sent to a land fill). The location of the remaining sources is not known. Generally, the sources whose whereabouts are unknown were small. There were however three reports of lost (or misplaced) sources containing 1 Ci or more:

Cs-137 - Kansas University Medical Center reported the loss of 5 Ci Cs-137 source.

Am-241 - Dresser Industries reported that a 4.8 Ci Am-241 source could not be located. It is believed that the source is still in the licensee's possession.

Kr-85 - Sweetheart Products Group reported the loss of a 1 Ci Kr-85 source on a common carrier.

4.1.4 Leaking or Contaminated Sources

Certain licensees are required to leak test sources and to report leaking sources under 10 CFR 34.25; others are required to leak test sources and to report leaking sources as a license condition. In both cases, a removable contamination exceeding the most common test limit for removable contamination (0.005 microcuries) is considered evidence of leakage, and must be reported to the NRC.

Twenty-one events of leaking or contaminated sources occurred during 1986. None of the events resulted in a radiation overexposure. The isotopic sources

found to be leaking or contaminated contained americium, cesium, cobalt, tritium, iodine, nickel, and technetium.

Most events were reports of small, individual sources found to be leaking or contaminated. One source leakage event was attributed to damage to sources during cleaning; another event was attributed to packaging; and a third event concerned damage to stolen radiopharmaceutical packages. One sealed source was discovered to have removable contamination after a steam leak in the facility.

A preliminary overview of the reports shows that the only generic problem displayed in the events, in which sources were damaged by cleaning, has been resolved by the licensee.

4.1.5 Release of Materials

Twenty-two events occurred in 1986 involving the release of radioactive materials. Several of these had significant consequences:

- ° In an event at Sequoyah Fuels in January 1986, a cylinder filled with from 14 to 15 tons of hot liquid uranium hexafluoride (UF-6) ruptured, releasing a large fraction of the contents as UF-6 gas. The UF-6 hydrolyzed, forming uranyl fluoride, UF-2F-2, and hydrogen fluoride (HF, hydrofluoric acid). The cloud of UF-2F-2 and HF was dispersed by a 25 mph wind. A highway adjacent to the plant was closed for a period of time. (See Section 4.1.7.)
- ° In an event at Wright-Patterson Air Force Base, a building used to house low level radioactive waste became contaminated when an unlabeled waste drum was opened on two occasions to inspect its contents in an attempt to determine the origin of the waste. After the first occasion, which resulted in contamination of the building, the Air Force Radiological Assistance Team was activated and arrived, together with a private contractor, to assist in decontamination and repackaging of the waste. The same drum was again opened, and resulted in further release of radioactivity. One individual may have received an uptake (inhalation) of airborne americium-241 that exceeded the NRC limit of 3-8 nanocuries. The uptake is still being evaluated. Costs to date for decontamination, repackaging, and disposal are about \$500,000. The fiscal decision whether to decontaminate the building further or to dismantle and dispose of it is still open.
- ° In an event in North Dakota, a tractor-trailer truck carrying yellowcake had an accident with a train. The accident resulted in contamination by yellowcake. The State Department of Health supervised the cleanup.
- ° Several reports of krypton-85 releases were received in 1986, of which none had serious consequences. There were six reports of losses of krypton-85 from fine leak test devices. The devices were manufactured by two different firms. The leaks were attributed to various causes: piping (plumbing) leaks; failure of a ten-year-old logic printed circuit board; installation of a new electrical component that was apparently inadequately sized; leaking valves; and inadequate maintenance. In two events, an O-ring seal on the device contributed to or caused the leak.

4.1.6 Consumer Products

An additional category, "consumer products," was defined for the data base in 1985. These reports describe events in which radioactive material was found in, or had a reasonable probability for being introduced into, nonlicensed consumer products. Four reports of this nature were received in 1986:

- ° A technician at Oyster Creek possessed several one dollar bills contaminated with I-131. He had received the money from a bank the day before the event.
- ° Several (97) contaminated or potentially contaminated snubbers were shipped from Farley Nuclear Station by Alabama Power and Light from Alabama to California. Ten snubbers were not located.
- ° A Florida licensee, Florida Steel, found a small piece of radioactive material, a Sr/Y beta source, in an incoming shipment of scrap.
- ° About 700 gallons of masking tape adhesive were contaminated with radioactive material from a static elimination device. The licensee, an Agreement State licensee, stated that the adhesive would not be used in products for commercial distribution without authorization.

4.1.7 Fuel Cycle Facility Event Reports

There were 13 fuel cycle events entered into the nonreactor data base in 1986. On January 4, 1986, a large cylinder filled with molten uranium hexafluoride ruptured at the Sequoyah Fuels Corporation Facility in Gore, OK. The event was perhaps the most significant fuel cycle event in many years, and as such, resulted in an increased sensitivity to events involving UF-6. A total of five of the 13 fuel cycle events concerned UF-6-related events; seven other events occurred at fuel fabrication facilities, and one event occurred at a uranium mill.

UF-6-Related Events:

- ° UF-6 Release at Sequoyah Fuels Facility, Gore, OK - On Saturday, January 4, 1986, a large cylinder of molten uranium hexafluoride ruptured while it was being heated in a steam chest. The force of the explosion damaged the steam chest enclosure. The escaping UF-6 rapidly reacted with moisture in the air to form uranyl fluoride and hydrofluoric acid. The resulting vapor cloud of these materials was carried south by southeast by a wind gusting to 25 mph. The cloud enveloped the process building, and the acidic vapor caused the death of an operator who was working approximately 70 feet from the cylinder. The vapor was drawn into the plant ventilation system. Approximately 40 workers in the building evacuated to an upwind location on site, some passing through the cloud. Plant personnel manned water hoses with fog nozzles in an attempt to suppress further airborne release of material.

Notification of NRC and civil authorities is estimated to have begun within 10 minutes. The injured workers were transported by fellow workers to nearby hospitals for treatment. The general population downwind was contacted and advised to evacuate and report to hospitals for examination.

Various local, State, and Federal officials were notified by the licensee from the corporate office in Oklahoma City. Residents downwind of the site were contacted by licensee employees and advised to proceed to a local hospital. The general public was notified by local radio.

Within an hour and a half, radiological surveys began both on and off the site. State, Federal, and company officials arrived at the site during the ensuing hours, and recovery operations began.

The NRC initiated an Augmented Investigation Team (AIT) inspection (the first such AIT for a fuel cycle event). The AIT report, "Rupture of Model 48Y UF-6 Cylinder and Release of Uranium Hexafluoride" (NUREG-1179), concluded that the cylinder was not defective but failed because of stress caused by hydraulic pressure that resulted from the expansion of the UF-6 in the cylinder when it was heated.

There were two major causes of the event:

- The physical equipment and facilities used for filling and weighing UF-6 cylinders were inappropriate for safe use with 14-ton cylinders.
- The training of workers in operating procedures and ensuring the implementation of these procedures were not carried out effectively.

In addition to forming an AIT, the NRC undertook three other major actions to appraise the significance of the event at Sequoyah Fuels:

- An Ad Hoc Interagency Public Health Assessment Task Force was established. The Task Force published a report, "Assessment of the Public Health Impact from the Accidental Release of UF-6 at the Sequoyah Fuels Corporation Facility at Gore, Oklahoma" (NUREG-1189), documenting their assessments.
- A Lessons-Learned Task Force (LLTF) was established. Their findings were furnished in the report, "Release of UF-6 from a Ruptured Model 48Y Cylinder at Sequoyah Fuels Corporation Facility: Lessons-Learned Report" (NUREG-1198).
- A Study Group was established to review the regulation of nuclear materials safety by the NRC. Their report, "Materials Safety Regulation Review Study Group Report," was published for public comment in the Federal Register on December 17, 1986.

Recommendations in the LLTF and the Study Group reports, as well as the public comments on the Study Group report are currently under review within the NRC.

- ° UF-6 Cylinder Deformation at Allied Chemical Company, Metropolis, IL - Following the Sequoyah event, the NRC Region III Office questioned Allied about cylinder overfills at its facility in Metropolis, IL. The NRC had been called by a newspaper reporter asking about a 1984 overfill at Allied. Allied, on questioning by NRC, stated that an overfilled cylinder had been heated in December 1984 with resulting damage to the cylinder (cracked stiffening rings; deformed wall). There were no releases of UF-6

during the event. NRC held that the event was reportable under Chapter 10 CFR Part 20. (Allied informed the NRC of an additional UF-6 cylinder overfilling event of March 23, 1986. The excess UF-6 was removed from the cylinder without heating.)

- Leaking Valve on UF-6 Cylinder - Allied Chemical Company filed a report of a leaking valve on a UF-6 cylinder undergoing an air pressure test.
- Transportation Accident on August 7, 1986 - A UF-6 cylinder being shipped from Babcock and Wilcox Company (Lynchburg, VA) to DOE broke free of the transport truck and fell on the street. Although there was superficial damage to the cylinder, the vent valve cover and plugs were intact and all seals were in place.
- UF-6 Pigtail Failure at Nuclear Fuel Services - On September 14, 1986, a pigtail between a heated UF-6 cylinder (in a glove box) and process equipment failed, releasing UF-6 to the glove box and the ventilation system. The calculated dose commitment to a child would be 0.02 mrem whole body exposure (at 250 meters). The pigtail was of a design that was not normally used.

Other Fuel Cycle Events:

- Milling - Plateau Resources Ltd. calculated that exposures of two employees working in the yellowcake packaging area exceeded 40 MPC hours.
- Uranium Fuel Fabrication -
 - At the Babcock and Wilcox Navy Nuclear Fuels Division facility, there was an accumulation of uranium in ventilation ducts that exceeded 75% of the critical mass value of 0.63 kg U-235. The accumulation of uranium was not adequately controlled.
 - During a routine survey of an unrestricted area at the Babcock and Wilcox Research Center, it was discovered to be contaminated. Signs were posted, and a fence was extended to encompass the restricted area.
 - A finishing unit at Nuclear Fuel Services (NFS) became pressurized, resulting in the release of radioactivity into the processing area. No overexposures resulted from the release.
 - NFS personnel were washing contamination off their hands in a lunchroom sink.
 - There was an elevated uranium concentration in sludge at a waste treatment plant in Erwin, TN. It was determined that the uranium comes largely from releases at an NFS onsite laundry facility. NFS was to take action to reduce the releases.
 - There was a hydrogen fire at NFS occurred on November 5, 1986. Reserve tubes of hydrogen located outside of the process buildings.

The licensee declared an unusual event and implemented its Radiological Contingency Plan. Most plant areas were temporarily evacuated. No radioactive materials were involved.

- General Electric Company reported that defective installation of high efficiency particulate air filters occurred in 1985. Installation procedures were deficient.

4.1.8 Industrial Radiography

Nineteen 1986 events involved radiography. One event occurred at a fixed radiography site and 18 occurred at remote (field) radiography sites or during transportation.

Seventeen of the reports concerned overexposure or potential overexposure events, and have been discussed previously. The number of radiography events does not differ substantially from the number of events reported during prior years.

4.1.9 Manufacturing and Distribution

Thirty-four 1986 events were identified from the program code of the licensee as being associated with manufacturing and distribution. These licensees have no unique reporting requirements for events involving health and safety, unless the requirements are incorporated into a license condition or an order. None of the events was significant.

4.1.10 Gauges/Measuring Systems

Holders of specific licenses to possess gauges are required to report failures of or damage to shielding; on/off mechanisms, or indicators of the gauge; or detection of removable contamination on the gauge. In addition, these licensees must make reports required pursuant to 10 CFR Part 20 (lost, or stolen materials, releases of material, etc.). Gauge licensees that submitted reports of events occurring in 1986 were identified by the program codes of the licensee. None of the events was by itself significant.

4.1.11 Abnormal Occurrences

In the Report to Congress on Abnormal Occurrences for the first half of 1986, four events at NRC licensees and five events at Agreement State licensees were determined to be abnormal occurrences (see Appendix A for listing). All of the events in the Agreement States occurred in 1985 or earlier.

The abnormal occurrences at NRC licensees involved:

- ° The rupture of a UF-6 cylinder and subsequent release of material at Sequoyah Fuels Corporation, Sequoyah, OK, facility.
- ° Two overexposures, one to a member of the public, and the other to a licensee employee.
- ° Breakdown of management controls at an irradiator facility.

With the exception of the occurrence at Sequoyah Fuels, the events were discovered as the result of routine inspections. The overexposures fell into the moderate range; i.e., they were less than 25 rem whole body.

The five abnormal occurrences at Agreement State licensees involved:

- ° Three exposures of industrial radiographers and an assistant radiographer (one of the overexposures was discussed in the 1985 Nonreactor Event Report, AEOD/N601).
- ° Contamination of a scrap steel facility (discussed in the 1985 Nonreactor Event Report); and
- ° Release of Kr-85 from a Trio-Tech "Tracer-Flo" system, resulting from a failure in the machines logic board. No overexposures were caused by the release.

The two radiography events in Agreement States that were not discussed in earlier NREER reports resulted in: a hand exposure of about 29,000 rem and a whole body exposure of 47 rem; and a hand exposure of about 2000 rad and a whole body dose of 6 rad.

Three nonreactor events were recommended to the Commission as abnormal occurrences during the second half of 1986. The events, all of which occurred at NRC licensees, include:

- ° The event at Wright-Patterson Air Force Base that resulted in contamination of a building when a waste drum was opened on two separate occasions.
- ° Suspension of a license of a radiography and teletherapy servicing firm.
- ° An order removing an officer of a radiography licensee from any assignment or position related to NRC-licensed activities.

The event at Wright-Patterson was reported to the NRC. The other events, not incorporated into the Nonreactor Event file, resulted from investigation of allegations (suspension of license) and observations by an NRC inspector during a routine inspection (removal of official).

4.1.12 Conclusions

As in prior years, most 1986 nonreactor events concerned reports of modest overexposures, lost or abandoned sources, or leaking sources. For these types of events, the 1986 data do not differ substantially from the same types of events reported in prior years.

The UF-6 cylinder rupture in January 1986 at Sequoyah Fuels was the most significant event reported by fuel cycle or material licensees for many years. That event resulted in a substantial effort by NRC to review its licensing and inspection activities in the nonreactor area, and to address the lessons taught by the event.

Another item of interest occurred, the contamination of a building at Wright-Patterson Air Force Base. The opening and dumping of an unlabeled waste

drum to identify its origin resulted in contamination of a building. Recovery costs of at least \$500,000 are estimated.

4.2 Medical Misadministration Events

The NRC regulates certain aspects of the uses of reactor-produced radioisotopes in nuclear medicine and therapeutic radiology. Certain diagnostic and therapy misadministrations must be reported to the NRC in accordance with 10 CFR 35.41 through 10 CFR 35.43. Diagnostic misadministration, as used in NRC regulations, refers to the misadministration of radioisotopes in nuclear medicine studies such as brain scans and bone scans. Therapy misadministration, as used in NRC regulations, refers to the misadministration of radiation from cobalt-60 teletherapy or radioisotopes in radiation therapy.

The significance of any event stems from the potential impact of the event on public health and safety. One dimension of event risk is the frequency of the event; a second is the magnitude of the potential impact of the event. AEOD has used the data collected on misadministrations for 6 years (1981-1986) to estimate error rates for certain of these misadministration events.

Regarding the frequency of events over the 6-year period, there were 25 therapy misadministration reports that involved teletherapy machines. In these 25 events, a total of 76 patients were overtreated or undertreated. Using patient statistics from the "Patterns of Care" study of the American College of Radiology, the error rate per patient is estimated to be about 0.015%. For diagnostic misadministrations, there were about 2400 reported to NRC over this 6-year period. A recent study by the Technologist Section of the Society of Nuclear Medicine estimated that about 10 million diagnostic procedures are performed annually in the United States. In 1986, NRC regulated only 22 of the 50 states; it is therefore estimated that about 4 million procedures are performed annually by NRC licensees. The diagnostic error rate per procedure is estimated to be about 0.01%.

Regarding the magnitude of the potential or actual impact of the event, therapy misadministrations are associated with procedures in which large doses of radiation are administered to patients to achieve a therapeutic effect. Diagnostic misadministrations are associated with procedures designed to permit a diagnosis to be made with little exposure to the patient. An exception is a diagnostic procedure known as an iodine-131 whole body scan which is discussed in detail below.

Therapy misadministrations have larger potential impacts on the health of the patient than diagnostic misadministrations. Diagnostic misadministrations that result in the erroneous administration of an iodine-131 whole body scan can result in thyroid doses that are near the therapy range. Since both teletherapy misadministrations and diagnostic misadministrations have about the same estimated error rate, the therapy misadministrations and some iodine-131 misadministrations as a class appear to be individually and collectively more significant than diagnostic misadministrations. AEOD, therefore, reviews in detail therapy misadministration reports and diagnostic misadministration reports that involve the administration of therapy amounts of radioisotopes (e.g., 1-5 millicurie of iodine-131 administered for a whole body iodine scan). Most diagnostic misadministration reports are reviewed from a collective or statistical viewpoint.

4.2.1 Occurrences Reported for 1986

For this period, 369 of the approximately 2600 NRC licensees authorized to perform nuclear medicine studies or radiation therapy reported one or more misadministrations, a total of 446 reports involving 498 patients (eight were therapy patients). Of the 446 reports of misadministrations, 438 (98%) reported diagnostic misadministrations, and eight (1%) reported therapy misadministrations.

4.2.2 Therapy Misadministrations

Six therapy misadministrations were reported in 1986. Three of the misadministrations involved teletherapy, one involved brachytherapy and two involved radiopharmaceutical therapy. The type and probable cause of the misadministrations is shown below:

Therapy Misadministrations Reported to NRC for 1986

Type of Procedure/Cause of Misadministration

Teletherapy

Error in dose calculations	1
Wrong patient was selected for treatment	1
Radiation administered from the wrong source (cobalt-60 instead of linear accelerator)	1
Source head rotation switch left in rotation mode	1

Brachytherapy

Error in dose calculations	1
Wrong activity sources loaded into source application	1

Radiopharmaceutical Therapy

Misunderstanding of verbal order	1
Failure to properly screen a patient for pregnancy	1*

While one of the teletherapy misadministrations, the two brachytherapy misadministrations, and one of the radiopharmaceutical therapy misadministrations were ascribed to causes previously identified in the AEOD report on therapy misadministrations (AEOD/C505, "Therapy Misadministrations Reported to NRC Pursuant to 10 CFR 35.42"), the causes of three of the misadministrations were new (i.e., not previously identified as causing a misadministration).

The four new causes were: (1) the wrong patient was selected for treatment; (2) a teletherapy treatment order using a linear accelerator was misunderstood as an order for a cobalt-60 machine; (3) the source head rotation switch on a cobalt-60 machine was inadvertently left in the rotation mode position; and (4) a patient was not properly screened for pregnancy. Although these causes are

*This event was reported by a licensee as a possible misadministration, but does not seem to fit the criteria of the regulations. It appears that the regulations did not address this type of event. It has been included because it seems to AEOD to be as significant as the other therapy misadministrations.

different from previously identified causes, the general conclusion drawn from the analysis of previous misadministration reports (i.e., that the occurrence of therapy misadministrations can be reduced by improvements in licensee quality assurance procedures) appears to apply equally to preventing misadministrations from the four newly identified causes.

4.2.3 Diagnostic Misadministrations

This section discusses the 1986 experience in terms of the number of diagnostic misadministrations by the types and causes of the misadministrations. Of the 438 total reports, 337 involved the administration of the wrong radiopharmaceutical to a patient and 80 involved the administration of a radiopharmaceutical to the wrong patient (95% of the reported misadministrations were of these two types). The remaining diagnostic misadministrations involved eight reports of the wrong route of administration, and 13 reports in which the diagnostic dose of a radiopharmaceutical differed from the prescribed dose by greater than 50%. The number of misadministrations reported for 1986 was about 13% higher than the average of the number of reports received for the last 2 years. However, because of the expected variations in the rate of occurrence of diagnostic misadministrations, we are not able to draw any conclusions from the data for the higher reporting rate. The types and causes of the diagnostic misadministrations were about the same as reported in 1985.

Effectively, all of the causes for these types of misadministrations involved human error. With regard to administration of the wrong radiopharmaceutical, the data show that 34 of the 337 events (10%) resulted from receipt of mislabeled doses from a radiopharmacy. In the remainder of the events, 30 (9%) resulted from misinterpretation of the physician's order, and errors in the preparation or delivery of doses accounted for 175 (52%). Another 98 reports (about 29%), had other causes or contained inadequate information from which to assign a cause.

The dominant causes for the wrong patient events were: the wrong patient's name on the requisition, 17 events (21%); the wrong patient was delivered to the nuclear medicine department, 18 events (23%); a failure to correlate the patient's identification with the study, 11 events (14%); and the patient answering to the wrong name, 13 events (16%). Relatively simple quality assurance procedures (checking the patient's identification against the study and the patient's medical history; asking the patient to state his name) might reduce the frequency of these events.

The remaining two types of diagnostic misadministrations, excess dose and wrong route of administration, had diverse causes.

4.2.3.1 Iodine Misadministrations

Although most of the diagnostic misadministrations involved the administration of the wrong technetium-99m compound to a patient or the administration of the technetium-99m compound to the wrong patient, five diagnostic misadministrations involved therapy doses of iodine-131 to patients. These misadministrations typically involved events in which the technologist performed a whole

body iodine scan on the patient, where the referring physician ordered a thyroid uptake or scan. The dose for whole body iodine scans is typically 1 to 5 millicuries; the typical dose for thyroid uptake study or scan is 30 microcuries of iodine-131, or 5 to 10 millicuries of technetium-99m. The "iodine-131 whole body scan" is the only diagnostic study where this large amount of iodine-131 is used.

The radiation dose to a patient is significantly higher if he is administered therapy doses of iodine-131 instead of the prescribed diagnostic dose of a radiopharmaceutical. For example, a patient prescribed a 5-millicurie dose of technetium-99m for a "thyroid scan" would receive 0.7 rad to the thyroid, whereas a 1-millicurie dose of iodine-131 can produce a dose of 4000-9000 rad to the thyroid.

During 1986, AEOD undertook an engineering evaluation of diagnostic misadministrations that involved administering therapy amounts of iodine-131 to patients. Although the study had not been issued as final at the end of CY 1986, preliminary findings included the following:

- (1) The direct causes of ten of the 14 reported iodine misadministrations (71%) were ascribed to either the physician's order being misinterpreted by or miscommunicated to the technologist (seven cases), or the technologist not knowing the correct dosage to administer for thyroid scan procedures that involved scanning the chest area (three cases).
- (2) Causal factors associated with the occurrences of the misadministrations appeared to include:
 - Use of verbal orders for nuclear medicine studies.
 - Use of similar terms by referring physicians and licensees to refer to different procedures.
 - Lack of technologist training.
 - Lack of procedures.
 - Failure of technologist to follow procedure.
- (3) The underlying cause of 11 of 14 (79%) of the misadministrations appears to have been a lack of licensee control over the administration of millicurie amounts of iodine-131 to patients. These 11 misadministrations could likely have been prevented, despite the errors that led to the misadministrations, if the prescription for the iodine-131 dosage had been verified for each patient before the iodine-131 was administered to the patient.

4.2.4 Abnormal Occurrences

Four misadministration events were included in the two Abnormal Occurrence Reports to Congress that were published for the first half of 1986; and five misadministration events have been recommended to the Commission as abnormal occurrences for the last half of 1986 (one of these latter events occurred in 1984 and one occurred in an Agreement State). The events deemed to be sufficiently serious to be classified as abnormal occurrences include five therapy misadministrations (two teletherapy, two brachytherapy, and one radiopharmaceutical therapy) and four diagnostic misadministrations. The diagnostic

misadministrations each involved the administration of a therapeutic amount of iodine to a patient.

4.2.5 Proposed and Pending Regulatory Changes Involving Misadministrations

The Commission's purpose in requiring the submittal of misadministration reports to the NRC is to verify that the causes are properly identified and that licensees implement appropriate corrective actions to prevent recurrence. If potential generic problems are identified, the Commission notifies other licensees of the generic problem or concerns and assesses the need for additional actions (e.g., changes in regulations to reduce the occurrence of similar and perhaps more serious events).

In this regard, as a result of the findings in AEOD case study AEOD/C505, "Therapy Misadministrations Reported to the NRC Pursuant to 10 CFR 35.42," and the findings from the investigation (by Regional Offices) of several recent misadministrations, the Commission directed NMSS to initiate rulemaking to require radiotherapy facilities to have quality assurance programs to, among other things, ensure the accuracy of patient doses. An ANPRM for this rulemaking is expected to be published in early 1986.

In addition, in late 1986 AEOD informed NMSS that AEOD was preparing to issue an engineering evaluation report that could be related to the ANPRM and proposed rule on quality assurance for radiotherapy facilities being developed by NMSS. The study would document (see Section 4.2.3.1) a review of diagnostic misadministrations that involved the administration of millicurie amounts of iodine-131 to patients.

4.2.6 Conclusions

Eight therapy misadministrations were reported in 1986, a rate not too different from prior years. Both the teletherapy and the brachytherapy misadministrations that occurred in 1986 might have been prevented by quality assurance procedures directed to verifying dose calculations, type of treatment, and patient identification.

Essentially all of the diagnostic misadministrations for 1986 involved either the administration of the wrong radiopharmaceutical or the administration of a radiopharmaceutical to the wrong patient. The type and cause of diagnostic misadministrations are about the same as reported for 1985. The causes reported by licensees are generally the same as have been reported in the past; that is, simple errors associated with (1) preparation of radiopharmaceuticals, (2) processing nuclear medicine requisitions, and (3) patient identification. An additional cause for misadministrations involving the administration of millicurie amounts of iodine to patients was the failure of licensees to exercise adequate control over the administration of millicurie amounts of iodine-131 to patients.

5. SUMMARY OF AEOD ACTIVITIES

An overall summary of AEOD reports issued in 1986 and in prior years is shown below:

<u>Summary of AEOD Reports</u> <u>Issued January 1, 1986 through December 31, 1986</u>	
° Case Studies	5 issued
° Special Studies	3 issued
° Trend and Pattern Studies	4 issued
° Reactor Engineering Evaluations	13 issued
° Reactor Technical Reviews	12 issued
° Nonreactor Engineering Evaluations	2 issued
° IIP Reports:	
- IIT Investigation Manual	1 issued
- IIT Investigation Manual Revision	1 issued
- IIT Investigation Report	2 issued

	<u>Summary by Year</u>						
	<u>1980</u>	<u>1981</u>	<u>1982</u>	<u>1983</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>
° Case Studies	5	5	6	2	5	5	5
° Engineering Evaluations and Technical Reviews	20	33	57	68	51	31	26
° Special Study and Trends and Patterns Reports	0	0	0	1	12	7	7
° Nonreactor Engineering Evaluations and Technical Reviews	0	8	10	14	4	3	2
° IIP Reports	--	--	--	--	--	1	4

Appendix B provides listings, by year, of all AEOD reports issued from 1980 through December 1986, including date published, title, and author.

5.1 Reactor Operations Analysis Branch (ROAB)

The Reactor Operations Analysis Branch (ROAB) performs the major technical AEOD studies on individual events and potential generic concerns. ROAB provides strong engineering and systems capabilities for the review of operational events involving U.S. and foreign commercial LWR plants. The operating experience of Fort St. Vrain, the only U.S. commercial gas-cooled reactor, is reviewed for ROAB under contract by Oak Ridge National Laboratory. ROAB is responsible for screening LERs and other pertinent operating information; identifying events involving particular safety significance; conducting in-depth engineering evaluations and case studies as warranted; and formulating appropriate recommendations for action by other NRC offices. The full scope of ROAB responsibilities is discussed below.

REACTOR OPERATIONS ANALYSIS BRANCH
STUART D. RUBIN, CHIEF

Responsibilities and Work Products

- 5.1.1 Systematically screen LERs and other pertinent domestic and foreign event reports and determine their significance
- 5.1.2 Analyze and evaluate selected individual events and potentially generic safety concerns associated with operational experience
- 5.1.3 Request followup NRC actions including:
 - Recommendations to address potentially significant safety concerns
 - Suggestions for feedback of important lessons learned
- 5.1.4 Document independent technical assessments including:
 - Case study reports (including a peer review process)
 - Special study reports
 - Engineering evaluation reports
 - Technical review reports
 - Memoranda
- 5.1.5 Implement the Memorandum of Agreement with INPO
- 5.1.6 Provide operational experience perspective and input to related agency activities

During this reporting period, a number of significant accomplishments and studies were completed. These are summarized below for each of the individual activities identified above.

5.1.1 Systematically Screen LERs and Other Pertinent Event Reports and Determine Their Significance

ROAB screens each LER and other pertinent event reports to identify those events or situations that warrant additional analysis and evaluation. The ROAB screening process is controlled by means of a written office procedure. From this process, ROAB determines whether: (1) further engineering review of the event by AEOD or others is warranted; (2) the event meets the criteria established for abnormal occurrence reporting; (3) the event meets the criteria established for reporting to the Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA) Incident Reporting Systems; and/or (4) the event should be described in the bimonthly AEOD publication, Power Reactor Events (see Section 5.2.4).

The fundamental objectives of the ROAB screening process are to identify and isolate precursor events and other situations where the margin of safety has been significantly degraded, and to identify from the operational experience generic situations or concerns which may have potential safety significance. A precursor is considered an event that could have been potentially serious if plant conditions, personnel actions, or equipment failure had been slightly different. Screening accomplishments during this reporting period include:

- ° A total of about 3000 LERs were screened by ROAB. Of these, approximately 150 LERs were judged to be significant, warranting additional NRC attention. Of these significant LERs, about 75% were related to generic issues or to matters being actively pursued by NRR, IR, or RES. The remainder (about 40 events) were designated by ROAB for further analysis and evaluation by AEOD.

In addition to U.S. experience, ROAB routinely reviews foreign operating experience reports. During this reporting period, 61 NEA/IRS reports and 31 IAEA reports were screened. Additionally, approximately 100 foreign reports obtained through bilateral agreements were reviewed.

Other U.S. operational information reviewed by ROAB during this reporting period included approximately 2500 regional inspection reports, 350 Part 21 reports, 4200 licensee 50.72 report summaries, 470 preliminary notification reports, and 50 INPO documents.

5.1.2 Analyze and Evaluate Selected Individual Events and Potentially Generic Safety Concerns

During this reporting period, ROAB completed one preliminary and four final case studies, one special report, 13 engineering evaluations, and 12 technical review reports. Representative studies are discussed below, a complete listing of completed studies is provided in Section 5.1.4. (Section 5.1.3 provides examples of follow-up NRC actions recommended by ROAB in the case studies discussed below and in selected engineering evaluations.)

- ° The AEOD preliminary case study on "Air Systems Problems at U.S. Light Water Reactors" was issued for peer review in December 1986. The report

provides a detailed, comprehensive and systematic analysis and evaluation of the potential safety implications associated with air system problems reported by domestic LWR licensees. The analysis of the operational data focuses on degraded air systems and the attendant vulnerability of safety-related equipment to common mode failures caused by air system degradations. The report analyzes the data for trends and patterns of the underlying causes and consequences, and evaluates the safety importance of the problem in terms of both risk assessments, and cost/benefit studies. The major finding was that the root cause of most air systems problems is traceable to either design or management deficiencies. The deficiencies appear to reflect a lack of sufficient regulatory requirements and review and the view by many applicants and licensees that air systems are not highly important to plant safety. The study presents several preliminary recommendations to reduce reactor accident risk, enhance safety and improve plant performance, including: (1) ensuring that air systems quality meets the requirements specified by the manufacturers of the plants' air-operated equipment; (2) ensuring adequate operator response by formulating and implementing anticipated transient and system recovery procedures for loss-of-air events; (3) improving training programs to ensure that plant operations and maintenance personnel are sufficiently sensitized to the importance of air systems and the potential vulnerability of safety-related equipment served by the air systems to common mode failures; (4) confirming the adequacy and reliability of safety-related backup air accumulators; and (5) verifying predicted equipment response to gradual losses of air by testing to ensure that such losses do not result in events which fall outside FSAR analysis assumptions. Following peer review, the final case study was issued in March 1987.

- Case study AEOD/C605, "Operational Experience Involving Losses of Electrical Inverters," was issued in December 1986. The study analyzed operational experiences involving losses of electrical inverters. Analysis of the data indicates that inverter loss events may be attributed to two causes: component failures and personnel actions. A major contributing factor and/or cause for inverter loss events was found to be incompatibility between actual plant service conditions and design service conditions for inverters. Contributing factors and/or causes for inverter losses were due to incorrect personnel actions which were traced to inadequacies in planning for maintenance and testing activities, and deficiencies in personnel training. The case study presented several recommendations to improve the operational performance reliability of inverters and related areas, including: issuance of an IE information notice focusing on the identified causes for occurrences of inverter loss events attributed to component failures and personnel actions; review of the technical specifications action statements addressing inverters and/or attendant bases for comparable plant designs to ensure consistency; and reassessment of a portion of the electrical circuitry associated with the Westinghouse Solid State Protection System. The report was forwarded to IE and NRR for review and appropriate followup regulatory or industry actions.
- Case study AEOD/C604, "Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems," was issued in December 1986. The report documents the analysis and evaluation

of four events involving the failures of solid state electronic components in safety-related instrument and control systems due to overheating caused by inadequate cabinet cooling. These events occurred at the McGuire 1, Davis-Besse, Summer 1, and Palo Verde 1 nuclear plants. Overheating of electronic components was found to raise two concerns: (1) decreased reliability of electronic components due to increased component failure rate, and (2) potential common mode failure of redundant safety-related instruments due to extended loss of normal cooling air flow to the instrument cabinets. Based on the study findings and conclusions, recommendations to address the significant safety concerns are presented in the report. The study recommendations include the need to: (1) establish emergency procedures and operator training to cope with a loss of cooling to instrument cabinets; (2) monitor actual local temperature conditions in the instrument cabinets; (3) provide technical specification requirements for operability of the control room cooling and ventilation systems which reflect the actual local temperature in the instrument cabinets; and (4) consider this issue in the plant-specific evaluation of the station blackout issue (USI A-44). The report was forwarded to NRR for review and appropriate followup action.

- Case study AEOD/C602, "Operational Experience Involving Turbine Overspeed Trips," was issued in August 1986. The study reviewed and evaluated past operating experience involving overspeed trips of PWR auxiliary feedwater (AFW) turbine-driven pumps. The study found that the causes of AFW turbine overspeed trips are dominated by speed control problems associated with governors, and trip and reset problems associated with trip valves and overspeed trip mechanisms. These problems result primarily from inadequate performance by plant personnel, inadequate procedures, and insufficient system design considerations. The governor speed control problems are caused by: (1) slow response of the Woodward Model PG-PL governor, (2) entrapped oil in the speed setting cylinder of the Woodward Model PG-PL governor, (3) incorrect governor setting, and (4) water induction into the turbine. The trip and reset problems were found to stem from the complexity of reset operations and a lack of trip position indication. Recommendations for preventing or reducing the overspeed trip problems were also presented in the study. The report recommended issuance of an IE information notice to feed back the lessons learned in the study to all LWR licensees, and consideration of specific additional recommendations in the ongoing NRR programs for improving AFW system reliability (particularly in response to action items directed by the EDO in connection with the NRC staff investigations of the June 9, 1985 event at Davis-Besse). Following issuance of the case study report, IE issued Information Notice 86-14, Supplement 1, "Overspeed Trips of AFW, HPCI, and RCIC Turbines," on December 17, 1986. Additionally, NRR agreed to consider the study recommendations in the resolution of Generic Issues 122 and 124.
- The final AEOD case study on "A Review of Motor-Operated Valve Performance" was issued in December 1986 (AEOD/C603). The study reviewed nuclear plant operating experience involving motor-operated valves in all safety-related systems from 1978 through mid-1985. The study found that: (1) valve assembly failure to operate continues to be a safety concern with little improvement in performance or reliability over the study period, and (2) current programs and procedures at many plants need

improvement to assure that motor-operated valve assemblies will operate when needed (e.g., under credible accident conditions). The study report presented several recommendations, including: reiteration of the recommendations presented in previous AEOD reports on the subject of motor-operated valves (C203 and S503); development of procedures and diagnostic capability for determining the root cause of valve assembly failures; development of training programs for plant personnel; and expansion of the scope of IE Bulletin 85-03 to all safety-related motor-operated valve assemblies. Following issuance of the case study report, it was transmitted to the Nuclear Utilities Management and Resources Committee (NUMARC) with the request that they take the lead in developing and implementing an industry program to address the findings, conclusions, and recommendations presented in the study.

In addition to the above case studies, AEOD issued a number of other studies which identified potentially significant safety concerns. Some representative examples are as follows:

- ° Engineering evaluation AEOD/E606, "Loss of Safety Injection Capability at Indian Point Unit 2," was issued in May 1986. The report evaluated a December 28, 1984 event in which all three safety injection pumps were inoperable. The malfunction of the safety injection pumps was caused by boric acid crystallization and possibly gas binding. Although the issue of boric acid crystallization is not viewed as a broad generic problem, the report identified additional plants that have a similar design which could also be prone to similar failures. Subsequent to the event, the licensee removed the boron injection tank (BIT) which was the source of boron crystallization. Other similar plants have also removed the BIT and/or obtained approval from NRR to operate with reduced boric acid levels to prevent similar malfunctions. The report's findings were discussed in IE Information Notice 86-63, "Loss of Safety Injection Capability."
- ° Engineering evaluation AEOD/E609, "Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation," was issued in August 1986. The report analyzed and evaluated 11 operational events which occurred at nine boiling water reactors over a period of 4 years. The operational events were primarily caused by human errors associated with the operation of the residual heat removal (RHR) system in the shutdown cooling mode. The cause of these human errors can be traced to deficient procedures, improper or inadvertent actions, lack of knowledge or training, man/machine interface problems, cognitive errors, or maintenance errors. The need for manual operation of the RHR shutdown cooling valves, the elevational differences and interconnections between the RHR subsystems, and the absence of comprehensive valve interlocks also contributed to the occurrence of these operational events. In light of the frequency and significance of these events, the report suggested several relatively low cost measures which could be implemented to prevent their recurrence. The study suggestions are consistent with the program elements stated in the NRC Human Factors Program Plan in the training, procedures, man-machine interface, and human performance areas. The study was transmitted to NRR for review, and suggested followup action within their ongoing human factors program.

- Technical review AEOD/T605, "Failure of Main Steam Safety Valves (MSSVs) to Properly Reseat," was issued in June 1986. The review of an event that occurred at Oconee with other operating experience indicates that the potential for MSSVs failing to reseat and initiating an overcooling transient is higher for Babcock & Wilcox designed plants than other PWRs. However, the likelihood of such an overcooling event challenging the structural integrity of the pressure vessel was found to be small. The study concluded that the industry evaluation of ways to reduce the number of challenges to MSSVs and improve the reliability of the valves was an adequate response to these concerns.
- Engineering evaluation AEOD/E602, "Unexpected Criticality Due to Incorrect Calculation and Failure to Follow Procedures," was issued in January 1986. The evaluation addressed an event at Virgil C. Summer Unit 1 that occurred on February 28, 1985. A reactor trip occurred on a high positive flux rate trip during a reactor startup. The event was attributed to a number of causes. First, the licensed operator conducting the startup failed to adhere to applicable procedures in that criticality was not anticipated during control rod bank withdrawal, and an awareness of plant conditions was not maintained at all times. The second cause which contributed to the event was a lack of adequate guidance in the procedures used to calculate the estimated critical rod position and reference critical data. Finally, the last cause identified which could have contributed to the event was procedural inadequacy in the licensee's administration of the plant's on-the-job training program. Although the event was bounded by accident analyses and adequate core protection was maintained, a review of related operating experience indicates that licensee operator training programs may need to be reviewed in order to minimize the potential for premature criticality events.
- Engineering evaluation AEOD/E603, "Delayed Access to Safety-Related Areas During Plant Operation," was issued in February 1986. The study reviewed several events of delayed access to vital areas at nuclear power plants caused by security, radiological protection or administrative provisions in order to determine their underlying causes and impact on plant operational safety. None of the events occurred during an actual plant emergency and therefore, no immediate safety concerns were involved. Most of the events were also of sufficiently short duration that had they occurred during an actual emergency, timely operator actions likely could have been taken if needed inside the affected compartment to avoid a significant degradation of plant safety margins. However, an event at Limerick involved delayed local operator actions during a remote shutdown demonstration test because plant procedures did not include provisions for having a set of equipment or compartment keys available at the remote shutdown panel. This procedural deficiency has been addressed by improvements implemented in the planning and procedures for remote shutdown operations. The engineering evaluation report suggested issuance of an IE information notice to remind licensees of the need for adequate key availability during remote shutdown operations. Following issuance of the report, IE Information Notice 86-55, "Delayed Access to Safety-Related Areas and Equipment During Plant Emergency," was issued.
- Technical review report AEOD/T606, "Inadvertent Recirculation Actuation Signals at Combustion Engineering Plants," was issued in August 1986. The

study was prompted by the different system responses to an inadvertent recirculation actuation signal (RAS) at Arkansas Nuclear One Unit 2 and Calvert Cliffs Unit 2. The study found that at CE plants without anti-draindown check valves in the emergency core cooling system pump suction lines, a significant inventory of borated water could transfer from the reactor water tank (RWT) to the containment sump on an inadvertent RAS. Antidraindown check valves prevented a similar loss of RWT inventory at the other plant design. Because plant responses were so different, the safety aspects of a postulated safety injection actuation signal subsequent to an inadvertent RAS were analyzed for each plant design. The analysis showed that, for either plant design, adequate suction flow would be provided to the ECC pumps in event of coincident RAS and LOCA signal.

- ° Engineering evaluation AEOD/E604, "Spurious System Isolations Caused by the Panalarm Model 86 Thermocouple Monitor," was issued in March 1986. The report evaluated a number of high pressure coolant injection, reactor core isolation cooling, and reactor water cleanup system isolations caused by spurious trips of a Panalarm Model 86 thermocouple monitor. The study found that the elevated sensitivity of the Panalarm thermocouple monitor makes the instrument highly susceptible to spurious trips caused by momentary electrical disturbances. The spurious system isolations caused by the instrument were found to have potentially adverse impacts on system reliability, isolation valve operability, and the distractions presented to the plant operating personnel. The study found that a modification to the leak detection system trip circuitry at Duane Arnold appears to have proven successful in significantly reducing spurious system isolations caused by the thermocouple monitor. The study suggested that other licensees be informed of the spurious system isolations caused by the Model 86 thermocouple monitor and the modification implemented at Duane Arnold which appeared to be an effective corrective action. In accordance with the study suggestion, IE issued Information Notice 86-69, "Spurious System Isolations caused by the Panalarm Model 86 Thermocouple Monitor," on August 18, 1986.
- ° Engineering evaluation AEOD/E610, "Loss of LPCI Loop Selection Logic at Millstone-1," was issued in August 1986. The report evaluated an event at Millstone-1 in which three Model 288 Barton differential pressure (D/P) switches in the low pressure coolant injection (LPCI) loop selection logic were found in the tripped condition. Under certain pump flow conditions, improper operation of these switches could result in the LPCI system injecting into the broken recirculation loop following a postulated design basis loss-of-coolant accident. The study found that the seven BWRs still equipped with LPCI loop selection logic utilize this type of switch. The switch is also used in a number of safety-related applications at various other LWRs. The study concluded that Model 288 Barton D/P switches are susceptible to setpoint drift and this susceptibility appears to increase as the switches age. The study also concluded that other BWRs could be vulnerable to this type of LPCI system degradation due to the susceptibility of the switches to setpoint drift. Suggestions were made to issue an IE information notice discussing the Millstone-1 event and the susceptibility of the Barton switch to setpoint drift. The study also suggested that IE evaluate the adequacy of the manufacturer's surveillance and maintenance recommendations.

- Engineering evaluation AEOD/E611, "Seismic Anchors for Electrical and Control Panels," was issued in October 1986. The report evaluated recent LERs involving inadequacies in the seismic anchorage of safety-related electrical and control system equipment at older operating plants. The report details several instances of design and installation deficiencies found in the seismic anchorage of electrical and control panels at the Dresden and Quad Cities nuclear power plants. The report noted that these deficiencies represented a failure to properly implement seismic design requirements in contrast to inadequacies in the seismic requirements themselves, which was the principal concern of USI A-46. Even so, the study provides considerable support for the proposed resolution of USI A-46, "Seismic Qualification of Equipment in Operating Nuclear Power Plants." The principal suggestion made in the report was to inform those plants not included in the resolution of USI A-46 of the types of design and installation errors identified in the study. Following issuance of the report, a generic letter on USI A-46 was issued which referenced the events reviewed and the concerns raised in the engineering evaluation.
- Engineering evaluation AEOD/E607, "Degradation or Loss of Charging Systems with Swing Pump Designs," was issued in July 1986. The report identified situations involving functional degradation or loss of charging systems using swing pump designs. The degradations or system losses were due to design deficiencies in interlock circuitry or inadequate maintenance procedures. In view of these deficiencies the study concluded that these or similar problems could exist at other plants. As a result of this study, IE Information Notice 86-79, "Degradation or Loss of Charging Systems at PWR Nuclear Power Plants Using Swing-Pump Designs," was issued in July 1986.
- Engineering evaluation AEOD/E605, "Lightning Events at Nuclear Power Plants," was issued in April 1986. The report was based on the review of 62 event reports involving lightning strikes at U.S. nuclear plants in the period from 1981 to 1985. The events occurred at 30 plant sites and involved 32 reactor units. A comparison of the number of lightning events, the geographic location of the affected units and the annual lightning strike density at the location, found a direct correlation between the annual lightning strike density and the number of events reported. The reported systems affected were: (1) the offsite power system, (2) the safety-related instrumentation and control system, (3) the meteorological and weather systems, (4) the radiation, gas and effluent flow monitoring systems, and (5) the air intake tunnel halon system. The report concluded that although lightning strikes have adversely affected the operation of some nuclear plants, in most cases there has been no significant degradation of safety systems and minimal equipment damage. In particular cases where damage has been extensive or where failures caused by lightning strikes have been repetitive, the licensees have taken corrective actions to reduce the consequences of future strikes. Since IE Information Notice 85-86 had been issued on the subject of lightning strikes, the report suggested that no further actions be taken.

- Technical review AEOD/T608, "Hydrogen Fire and Failure of Detection System," was issued in November 1986. The report documents the review of an event at a foreign reactor which involved a leak in the unit's electric generator hydrogen system followed by an explosion and a fire. Due to a design deficiency in the fire detection system that allowed too many detectors to be installed in a detection loop, the fire detection system did not respond correctly during the event--instead of annunciating a fire, it indicated an "out of order" condition. The fire was detected promptly in spite of the detection system problem, and the plant was safely shut down. The review was initiated to assess whether such a fire detection system problem could exist at U.S. plants, since such a design deficiency could potentially prevent the detection system from performing its design function when needed. Based on a review of recent operational events in domestic plants involving fire protection and detection systems, such a design deficiency was not identified in U.S. nuclear plants. Although a review of detailed design drawings of fire detection systems installed in specific U.S. plants was not performed, the report concluded that a large fire, that renders the fire detection system incapable of functioning properly, remaining undetected is a low possibility.
- Technical review report AEOD/T612, "Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches," was issued in December 1986. The report addressed in detail three events at D.C. Cook and one event at Kewaunee where the ability of a safety system to automatically perform its designed function was compromised due to human error. Additionally, 55 similar incidents which occurred at U.S. operating plants between 1981 and 1985 were examined. Nearly 91% of these 55 events occurred at PWRs and the rest at BWRs. The systems most affected in these events were the safety injection system, the residual heat removal system and the containment spray system. The events at D.C. Cook and Kewaunee and other similar occurrences reviewed illustrate that human error can result in the degradation or complete loss of a safety system. These events emphasize the need for appropriate procedural controls, adequate communications between plant personnel, and independent verification of system lineup after maintenance or surveillance activities.
- Engineering evaluation AEOD/E612, "Emergency Diesel Generator Component Failures Due to Vibration," was issued in December 1986. The report evaluated eight events that involved cracking of small bore piping that resulted in inoperability of an onsite emergency diesel generator (EDG). The cracked lines were found in the EDG lube oil, fuel oil and cooling water systems. The piping cracks, which were caused by cyclic fatigue that resulted from engine-induced vibration, were not detected by either the inservice inspection or the preoperational testing program for the piping. The failures were discovered after the cracks propagated completely through the tube wall and fluid was observed leaking from the pipes. The study found that fatigue failures induced by steady-state operation, such as plant equipment or engine-induced vibration, are not normally analyzed in the original piping design. A more complete design review is typically done only after related problems are identified during plant operation. Often the problem is not detected until a leak occurs. Such leaks could result in the sudden disabling of the EDGs when needed

and could adversely affect a safe plant shutdown following of a loss of offsite power. The report suggested that IE issue an information notice to address the potential safety concern of fatigue cracks in those small bore lines associated with EDGs.

- Engineering evaluation AEOD/E613, "Localized Rod Cluster Control Assembly (RCCA) Wear at PWR Plants," was issued in December 1986. The report found that RCCA wear should be anticipated in Westinghouse designed PWRs. The degradation was related to: (1) wear during rod motion associated with startup, shutdown, reactor trip, and load following; (2) flow induced vibration leading to fretting wear between the rodlet and the guide card when RCCAs are at fixed positions for long periods of time; and (3) rodlet cracking (intergranular stress corrosion cracking) that appears related to absorber/clad interaction. The concerns raised by this study were forwarded to IE with a request for an IE information notice and to NRR with a request to review whether periodic inspections by licensees were needed.
- Engineering evaluation AEOD/E601, "Deficient Operator Actions Following Dual Functions Valve Failures," was issued in September 1986. The study evaluated events involving dual function valve failures at Dresden 3, Brunswick 1 and 2, and Peach Bottom 3. The study found that LWRs generally are equipped with a number of valves which perform both an emergency core cooling (or safety-related) function and a containment isolation function. However, operating experience shows that the proper and conservative operator action for a failure of one of these valves has not always been taken by the operating staff in a manner which is fully consistent with plant technical specifications. In each of the events reviewed, the operator's actions preserved one function of the valve but disabled the valve's alternate function. Furthermore, the operator's actions were taken without declaring the adversely affected function of the valve to be inoperable as required by the plant technical specifications. At Dresden 3, operability of an ECCS loop was preserved, but containment isolation capability was disabled. At Brunswick and Peach Bottom 3, containment integrity was maintained, but a safeguards train was disabled. The study also found that the affected valves were not included in the plant technical specification table of containment isolation valves. As a result, IE Information Notice 86-38, "Deficient Operator Actions Following Dual Function Valve Failures," was issued, and the Technical Specification Improvement Program adopted a suggestion to highlight these dual function valves in all plant technical specifications.
- Engineering evaluation AEOD/E604, "Spurious System Isolations Caused by the Panalarm Model 86 Thermocouple Monitor," was issued in March 1986. The report evaluated a number of high pressure coolant injection, reactor core isolation cooling, and reactor water cleanup system isolations caused by a spurious trip of a Panalarm Model 86 thermocouple monitor. The study found that the elevated sensitivity of the Panalarm thermocouple monitor makes the instrument highly susceptible to spurious trips caused by momentary electrical disturbances. The spurious system isolations caused by the instrument are undesirable because of the potentially adverse impacts on system reliability, isolation valve operability, and the distractions presented to the plant operating personnel. The study found

that a modification to the leak detection system trip circuitry at Duane Arnold appears to have proven successful in significantly reducing spurious system isolations caused by the thermocouple monitor. The study recommended that other licensees be informed of the spurious system isolations caused by the Model 86 thermocouple monitor and describes the modification implemented at Duane Arnold as a possible corrective action. IE issued Information Notice 86-69, "Spurious System Isolations caused by the Panalarm Model 86 Thermocouple Monitor," on August 18, 1986.

In addition, the following special study report was performed and issued by ROAB during this reporting period:

- ° Special study AEOD/S603, "Adequacy of the Scope of Bulletin 86-01," was issued in June 1986. This study evaluated a potential inadequacy in the scope of IE Bulletin 86-01, "Minimum Flow Logic Problems That Could Disable RHR Pumps," which had been issued in May 1986. Bulletin 86-01 was addressed to only BWR facilities. The AEOD study included information which strongly indicated that a problem similar to the BWR residual heat removal (RHR) minimum flow control logic single failure vulnerability identified at the Pilgrim station (the topic of the IE Bulletin) might also exist at a number of Westinghouse-designed PWRs. Documents from Wisconsin Electric and Carolina Power and Light described almost identical and analogous design deficiencies at Point Beach and H. B. Robinson. The deficiency at both plants involved a single failure that could cause the loss of the common minimum flow recirculation path for all of the safety injection (SI) pumps. The deficiency at Point Beach had also been the subject of a Part 21 report. Due to the potential safety significance which could be associated with a failure of the SI system at Westinghouse PWRs and the timeliness of IE Bulletin 86-01, the memorandum recommended that IE give immediate consideration to issuing either a supplemental or a separate IE bulletin to all Westinghouse PWR facilities concerning the SI recirculation path design deficiencies identified at Point Beach and H. B. Robinson. Subsequently, on October 8, 1986, IE issued Compliance Bulletin 86-03, "Potential Failure of Multiple ECCS Pumps due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line."

5.1.3 Request for Followup NRC Actions

The reports issued by ROAB during this reporting period included a number of formal recommendations for NRC program office followup actions related to potentially significant safety concerns or suggestions for feedback of important lessons learned from plant operating experiences. In addition, during this reporting period, a number of actions were initiated or completed by one or more NRC program offices as a result of recommendations or suggestions previously made by ROAB. Selected examples of recommendations or suggestions made by ROAB during this period, or actions taken by others as a result of earlier requests, are presented below (see Section 6 for a complete status report on all AEOD recommendations that were outstanding during 1986).

- ° Engineering evaluation AEOD/E602 was forwarded to NRR, Region II, and Region IV for information and review. As a result of this review, NRR performed a special evaluation of the operator licensing examinations conducted at Surry, and subsequently upgraded future examinations to place more emphasis on reactor start procedures.

- ° Engineering evaluation AEOD/E606 suggested that an IE information notice be issued to make the licensees of potentially affected plants more aware of the safety concerns which were raised by the Indian Point Unit-2 loss of all safety injection pumps due to boric acid crystallization. It was suggested that licensees of potentially affected plants be made aware of the safety concerns of boric acid crystallization, and emphasize the need to follow strict administrative controls to prevent similar occurrences until their boron injection tanks are removed. In response, IE Information Notice 86-63, "Loss of Safety Injection Capability," was issued.
- ° Engineering evaluation AEOD/E609 suggested that several effective and relatively low cost measures, developed by the study to reduce the likelihood of recurrence of inadvertent draining of the reactor vessel, be considered by NRR as part of their ongoing systematic assessment of human factors concerns. As a result, NRR agreed to give consideration to these measures as part of their ongoing evaluations in the human factors area.
- ° Engineering evaluation AEOD/E603 suggested that an IE information notice be issued on an event at Limerick where plant procedures did not include provisions for having a set of equipment or compartment keys available at the remote shutdown panel. As a result, IE Information Notice 86-55, "Delayed Access to Safety-Related Areas and Equipment During Plant Emergency," was issued on this subject.
- ° Engineering evaluation AEOD/E601 suggested that an IE information notice be issued on several events involving the failure of dual function valves. In each of the events reviewed, the operator's actions preserved one function of the valve, but disabled the valve's alternate function. As a result, IE Information Notice 86-38, "Deficient Operator Actions Following Dual Valve Failures," was issued and the technical specification improvement program included the suggestion that valves which perform both an ECCS and containment isolation function be highlighted in all plant technical specifications.
- ° Engineering evaluation AEOD/E604 suggested that an IE information notice be issued on operating events involving high pressure coolant injection, reactor core isolation cooling, and reactor water cleanup system isolations caused by spurious trips of Panalarm Model 86 thermocouple monitors. As a result, IE Information Notice 86-69, "Spurious System Isolation Caused by the Panalarm Model 86 thermocouple Monitor," was issued.
- ° Engineering evaluation AEOD/E610 suggested that an information notice be issued that describes the loss of LPCI selection logic which occurred at Millstone-1 and the susceptibility of the Barton Model 288 differential pressure (D/P) switches to setpoint drift. It was also suggested that the IE Vendor Inspection Branch evaluate the adequacy of the manufacturer's recommended surveillance and maintenance with regard to the intended usage of the Model 288 D/P switch in the LPCI loop selection logic. The Vendor Inspection Branch is investigating these concerns and IE will issue an information notice pending the results of its investigation. NRR plans to include information on the Barton Model 288 as part of its generic letter on differential switches.

- Engineering evaluation AEOD/E611 suggested: (1) issuance of an updated IE information notice to characterize the continuing deficiencies in seismic anchorage of critical safety equipment; (2) reconsideration of the limitation of the application of the resolution of USI A-46 to 70 older plants since the deficiencies identified in the LERs and in recent NRC inspections are caused by construction/quality assurance deficiencies and not design; and (3) preliminary walk-through reinspections to identify gross deficiencies, as indicated by the LERs, before the detailed reinspections and engineering analyses are conducted. In response to these suggestions, a generic letter for plants not included in the USI A-46 program was revised to address the study findings and conclusions.
- Special study AEOD/S603 alerted IE to design deficiencies identified at several Westinghouse-designed PWRs involving the potential loss of all safety injection pumps due to a single failure in the minimum flow recirculation path. As a result, IE issued IE Compliance Bulletin No. 86-03, "Potential Failure of Multiple ECCS Pumps due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line."
- Case study AEOD/C602 contained several recommended actions aimed at preventing or reducing the unwanted turbine overspeed trips identified in past operational events. Specifically, the report recommended that: (1) an IE information notice be issued to address the study findings; (2) measures be taken to alleviate the effect of slow response of the governor valve; (3) administrative controls be implemented or an additional device be added to prevent hydraulic fluid from entrapping in the speed setting cylinder; (4) additional review take place for the existing vendor supplied calibration procedures for governor speed setting to assure adequacy; (5) adequate provisions be provided for condensate removal from the steam supply line to prevent water induction into the turbine during startup; (6) the adequacy of the existing procedural instruction be assessed; (7) training programs be upgraded; and (8) position indication to minimize trip and reset problems be provided. In response to these recommendations, IE Information Notice 86-14, Supplement 1, "Overspeed Trips of AFW, HPCI and RCIC Turbines," was issued. Additionally, NRR agreed to consider the study recommendations in the resolutions of Generic Issues 122 and 124.
- Engineering evaluation AEOD/E607 suggested that an information notice be issued for the events, causes, and possible corrective actions associated with functional degradation or loss of charging systems using swing pump design. In response to this request, IE Information Notice 86-79, "Degradation of Loss of Charging Systems at PWR Nuclear Power Plants Using Swing-Pump Designs," was issued.
- Case study AEOD/C603 contained a number of recommendations associated with the need for high priority licensee action in several areas related to motor operated valve performance. The EDO subsequently sent a request to NUMARC to assume the lead in a concerted industry effort to address the AEOD recommendations.

5.1.4 Document Independent Technical Assessments

During 1986, the following studies were completed by ROAB personnel:

Reactor Case Study Reports - 1986

<u>No.</u>	<u>Subject</u>	<u>Date Issued</u>	<u>Author</u>
C602	Operational Experience Involving Turbine Overspeed Trips	8/20/86	C. Hsu
C603	A Review of Motor-Operated Valve Performance	12/5/86	E. Brown
C604	Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems	12/30/86	M. Chiramal
C605	Operational Experience Involving Losses of Electrical Inverters	12/31/86	F. Ashe

Special Study Reports - 1986

<u>No.</u>	<u>Subject</u>	<u>Date Issued</u>	<u>Author</u>
S603	Adequacy of the Scope of IE Bulletin 86-01	6/17/86	E. Leeds

Reactor Engineering Evaluation Reports - 1986

<u>No.</u>	<u>Subject</u>	<u>Date Issued</u>	<u>Author</u>
E601	Deficient Operator Actions Following Dual Function Valve Failures	1/9/86	E. Leeds
E602	Unexpected Criticality Due to Incorrect Calculation and Failure to Follow Procedures	1/15/86	R. Freeman
E603	Delayed Access to Safety-Related Areas During Plant Operation	1/19/86	T. Cintula
E604	Spurious System Isolations Caused by the Panalarm Model 86 Thermocouple Monitor	3/14/86	E. Leeds

<u>No.</u>	<u>Subject</u>	<u>Date Issued</u>	<u>Author</u>
E605	Lightning Events at Nuclear Power Plants	4/28/86	M. Chiramal
E606	Loss of Safety Injection Capability at Indian Point Unit 2	5/27/86	R. Tripathi
E514	Core Damage Precursor Event at Trojan (Revision 1)	5/27/86	D. Zukor
E607	Degradation of Loss of Charging Systems with Swing Pump Designs	7/3/86	F. Ashe
E608	Re-examination of Water Hammer Occurrences	7/14/86	E. Leeds
E609	Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation	8/8/86	P. Lam
E610	Loss of LPCI Loop Selection Logic at Millstone-1	8/14/86	E. Leeds
E611	Deficiencies in Seismic Anchorage for Electrical and Control Panels	10/16/86	N. Thomasson
E612	Emergency Diesel Generator Component Failures Due to Vibration	12/17/86	C. Hsu
E613	Localized Rod Cluster Control Assembly (RCCA) Wear at PWR Plants	12/23/86	E. Brown

Reactor Technical Review Reports - 1986

<u>No.</u>	<u>Subject</u>	<u>Date Issued</u>	<u>Author</u>
T601	Pressure Sensitive Temperature Switch Results in Spurious Actuation of Fire Suppression System	2/27/86	T. Cintula
T602	Emergency Diesel Generator Cooling Water system Design Deficiencies at Maine Yankee and Haddam Neck	4/29/86	E. Leeds
T603	Inadvertent Pump Suction Transfer and Potential Auxiliary Feedwater Pump Cavitation at Davis-Besse	5/6/86	R. Tripathi
T604	Events Resulting from Deficiencies in Labeling and Identification Systems	5/7/86	E. Trager
T605	Failure of Main Steam Safety Valves (MSSVs) to Properly Reseat	6/17/86	R. Freeman

<u>No.</u>	<u>Subject</u>	<u>Date Issued</u>	<u>Author</u>
T606	Inadvertent Recirculation Acutation Signals at Combustion Engineering Plants	8/7/86	T. Cintula
T607	Occurrence of Events Involving Wrong Units/Wrong Train/Wrong Component - Update Through June 1986	9/19/86	E. Trager
T608	Hydrogen Fire and Failure of Detection System	11/12/86	M. Chirama
T609	Foreign Material and Debris in Safety-Related Fluid Systems	12/16/86	E. Leeds
T610	ADS/RCIC System Interaction Events at River Bend-1	12/19/86	E. Leeds
T611	Denied Access due to Negative Room Pressure	12/19/86	T. Cintula
T612	Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches	12/31/86	R. Tripathi

5.1.5 Implement the Memorandum of Agreement with INPO

AEOD and INPO share results of completed studies related to review and assessment of operational data. Additionally, listings are periodically exchanged for studies planned or in progress. Periodic informal meetings are held to discuss concerns of mutual interest. During 1986, two such meetings were held (May 29 and November 19, 1986). In addition to these meetings, considerable informal discussion occurs between the ROAB staff and their INPO counterparts.

5.1.6 Provide Operational Experience Perspectives, Input, and Support to Related Agency Activities

ROAB is often requested to provide comments, perspectives and support on a number of technical issues and concerns and to serve on various task forces or teams. Such assignments are authorized when it is believed that AEOD can be responsive and the request offers a desirable method of communicating the lessons learned from operating experience and related AEOD studies. Examples of this type of activity during this reporting period, included:

- ° Participating in NRC's review of the B&W Owner's Group Reassessment Program
- ° Providing input to NRR on the resolution of Generic Issue 99 "RCS/RHR Suction Line Interlocks"
- ° Participating in the third regular bilateral NRC/MITI meeting on nuclear safety

- ° Presenting a paper at the International ANS/ENS Topical Meeting on Thermal Reactor Safety (February 1986)
- ° Submitting a paper for publication at the ANS Topical Meeting on Nuclear Power Plant Maintenance (March 1986)
- ° Assisting the IIT team on the Rancho Seco investigation
- ° Participating in the Augmented Inspection Team at the Pilgrim Nuclear Power Station
- ° Participating in the IAEA consultants' meeting on "IRS Generic Safety Issues Identification," and preparing a technical report on the subject
- ° Providing input to Commissioner's site visit briefings
- ° Providing input to NRR on the resolution of Generic Issue 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems"
- ° Participating in a team inspection of San Onofre Units 2 and 3 in connection with the ability of these units to safely respond to events that have occurred at other facilities.

5.2 Program Technology Branch (PTB)

PTB is responsible for a broad range of program, technical, and administrative activities within AEOD, as shown below.

PROGRAM TECHNOLOGY BRANCH
MARK H. WILLIAMS, CHIEF

Responsibilities and Work Products

- 5.2.1 Conduct a comprehensive and systematic trends and patterns program
- 5.2.2 Conduct Nuclear Plant Reliability Data System (NPRDS) evaluation program and coordinate NRC guidance and usage of the NPRD System
- 5.2.3 Formulate Abnormal Occurrence (AO) guidance and prepare AO reports
- 5.2.4 Issue operational data feedback reports, including bimonthly Power Reactor Events reports and monthly LER Compilation reports
- 5.2.5 Prepare reports on U.S. events to the Nuclear Energy Agency's Incident Reporting System (NEA-IRS) and provide support to the International Atomic Energy Agency's Incident Reporting System (IAEA-IRS)
- 5.2.6 Develop techniques to apply PRA perspectives to the screening and analysis of operational events

- 5.2.7 Develop and manage data bases, including:
 - Sequence Coding and Search System (SCSS)
 - Foreign Event File
- 5.2.8 Access INPO data files, including the NPRD System
- 5.2.9 Formulate and operate internal management information systems, including:
 - LER screening results (WAMS)
 - Resource accountability (TACS)
 - Incident Reporting System (IRS)

During this reporting period, a number of significant milestones and actions were accomplished. These are summarized below for each of the individual activities identified above.

5.2.1 Trends and Patterns Program Plan, FY 86-88

The AEOD Trends and Patterns Programs Plan for FY 86-88 was developed in 1985 and distributed to NRC Headquarters and Regional Offices for comment in January 1986. This plan describes the AEOD program for the periodic analysis of sets of operational event data reported by commercial power reactor licensees in the Licensee Event Reports (LERs) and to INPO's Nuclear Plant Reliability Data System (NPRDS). "Trends and patterns" is used to describe a program for analyzing incidents of low individual significance but which may be significant because of their frequency or distribution. Trends and patterns analysis usually assesses operational data with limited prior formulation of a concern, whereas an engineering review usually begins with the formulation of a specific concern.

The AEOD Trends and Patterns Program consists of three separate program elements:

- (1) Analysis of Event Data. Four indepth reports, each treating a different category of operational events covered by the requirements of 10 CFR 50.73 on reactor trips, ESF actuations, loss of system safety function, and technical specifications violations and shutdowns, and a fifth report reviewing the performance of recently licensed plants, will be issued covering the events from the most recent calendar year and comparing the latest results with that from earlier years. Further, each report will contain findings, conclusions, and recommendations for correcting problems found to be most significant. Some preliminary results of the 1986 studies are discussed in Section 3.
- (2) Analysis of Component Data. AEOD has developed and pilot tested a program for the systematic analysis of NPRDS data. The methods of analysis were tested, and the full-scale implementation of this program began in January 1986.

- (3) Performance Indicator Data. At present six indicators are being tracked by NRC's Performance Indicator Program. The basic input data for five of these indicators are developed by AEOD using contract assistance from INEL.

Results of analysis of event data:

The results and conclusions of the analysis of events documented in LERS were presented in the following reports issued in 1986:

- ° AEOD issued P603, "Trend and Patterns Analysis of Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Plants," in August 1986. This study covers the July-December 1984 time period. The analysis of 1985 data will be issued in the spring of 1987. The preliminary results from the 1986 data, now being compiled, are discussed in Section 3.7.
- ° AEOD issued report P602, "Trend and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1985," in August 1986. The 1986 report is scheduled for publication in June 1987.
- ° AEOD initiated trends and patterns analyses of losses of system safety function and violations of shutdowns required by technical specifications as reported in 1984 and 1985. The reports on these topics for 1985 will be issued in spring 1987.
- ° In August 1986, AEOD issued P604, "AEOD Evaluation of New Plant Experience," an analysis of operational data gathered in the first 2 years of licensed operation for new plants. A more in-depth study of root causes of events at new plants will be issued later in 1987.

Methods and results of analysis of component failure data:

In 1985, AEOD pilot tested a program to systematically analyze NPRDS data. The NPRD System is an industry-wide, voluntary reporting system for tracking the performance of selected systems and components at nuclear power plants. Since almost all U.S. plants in commercial operation supply detailed design data, operating characteristics, and performance data, NPRDS provides an extensive data source for analysis of operating experience. AEOD worked with INPO in developing a list of critical components, called "key components," which are considered to have the greatest impact on safety system availability and the occurrence of plant transients. The NPRDS Trends and Patterns Analysis Program focuses on these key components.

The statistical analysis of the key components utilizes four different methods using the time to failure, engineering information and failure counts in NPRDS:

- (1) Trend Analysis - Identifies the trend in the failure rate of key components over time and to determine if the rate is increasing or decreasing, and the significance of the trend.

- (2) The Page Procedure - Detects a shift in the failure rate of a particular key component above a preselected target or threshold failure rate.
- (3) Survival Analysis - Determines the factors (pedigree information) which appear to influence the failure pattern of key components and the failure rate of the component over the life of the component.
- (4) Reliability Modeling - Fits well-known reliability models to the failure patterns of key components; that is, to model the failure rate of the component over time, thus providing insight into failure causes and identifying outliers (components whose failure pattern suggests more detailed engineering analysis).

The results of these statistical analysis for each key component is then used by AEOD engineers as a starting point for an engineering evaluation to develop findings, conclusions, and recommendations for the selected key components. The full scale implementation of this program began in January 1986 with the analysis of failures of PWR main feedwater system components. The preliminary results of this analysis are discussed in Section 3.9. Three proprietary reports were prepared by INEL under contract to AEOD in 1986.

5.2.2 Conduct the NPRDS Evaluation Program and Coordinate NRC Guidance for NPRDS

As directed by the Commission, an NPRDS evaluation program continues to be implemented. Semiannual evaluation reports on NPRDS progress were forwarded in July 1986 (SECY-86-216) and February (SECY-87-54).

These reports highlighted the continued commitment of the industry to the improvement of the NPRDS. INPO's progress in the conversion of the NPRDS to an IBM based system will significantly improve the NPRDS capability and greatly enhance its usefulness. The reports also commended INPO for improvements in the number of plants participating, in the number of reports being submitted by the plants, and in the timeliness and quality of the data during the period covered by these reports, fourth quarter 1985 through third quarter 1986.

In particular, the following specific observations resulted from the staff's activities involving NPRDS:

- (1) All but one of the eligible plants submitted at least one failure report during the four quarters covered by these reports.
- (2) Over 4,000 failure reports were submitted by the plants in each quarter.
- (3) The timeliness of NPRD failure reporting for the quarters evaluated has improved compared to the same period in the preceding year. The reporting timeliness has stabilized over the years to the point where within 9 months of the end of the quarter approximately 75% of the NPRDS-reportable failures described in a sample of LERs were in the NPRDS data base; half of the failure reports are in NPRDS within about 6 months of the failure.

- (4) To assess the quality of the narrative information provided in NPRDS failure reports, a sample of reports was reviewed to determine if the text described the failure in sufficient detail such that system users could understand the failure events. For the four quarters evaluated, over 85% of the samples were rated as at least "probably adequate" (i.e., a knowledgeable person could understand the characteristics of the failure.)

Although these improvements are very notable, the data base completeness and quality continue to warrant attention. At present, its usefulness is limited by the resources required to verify and upgrade the data after it is taken from NPRDS. Particular observations in this regard are:

- ° The staff continues to assess whether its current program for measuring the level of NPRDS reporting, using the component failures reporting in LERs as a yardstick, accurately reflects the overall level of reporting. In an effort to improve the NPRDS, INPO has promoted the reporting of failures documented in LERs. Therefore, the reporting level of component failures from LERs that are also in NPRDS may be higher, perhaps significantly, than the overall reporting level. During the past evaluation period, AEOD tested two methods of measuring the overall completeness of NPRDS failure reporting. Although no conclusions can be drawn from these two checks as to the overall completeness of the data base, they do indicate that diversified measuring methods should be pursued.
- ° From an NRC user perspective, some difficulties have been found in using NPRDS data, such as inconsistencies in the engineering reports, missing engineering records, and errors in the data. User's comments are becoming an increasingly important part of the NPRDS evaluation/improvement process. We have been and will continue to provide INPO with this user feedback.
- ° Recent case studies that attempted to use NPRDS data found that narrative descriptions in the NPRDS failure reports were often too brief to adequately determine failure causes. For this reason our recent evaluations have also measured the percentage of failure reports that provided good root cause information. Few reports met this test.

Because of our concerns with the possible preferential reporting of LER documented failures and the problems in meeting NPRDS user needs, our future evaluations by the staff will be directed at determining failure reporting completeness and NPRDS appropriateness for the various end uses, including regulatory activities, and at stimulating needed improvements.

5.2.3 Prepare Abnormal Occurrence (AO) Reports

AEOD prepares the quarterly AO Reports to Congress (as well as the associated Federal Register Notices) and, after staff coordination, forwards them to the Commission. These reports, issued as the NUREG-0090 series, serve as a feedback of significant event information to Congress, government agencies, licensees, and the public. The reports are widely distributed (PDR, LPDRs, U.S. nuclear plants, Agreement States, other government agencies, and various foreign nations) and are available individually or on a subscription basis through the GPO sales program.

Three quarterly AO Reports to Congress were issued during calendar year 1986 (third and fourth quarter CY 85, and the first quarter CY 86). The three reports described 19 AOs at NRC licensees and four AOs at Agreement State licensees. The reports also described 11 additional events as other reports of interest. Of the 19 AOs at NRC licensees, seven occurred at nuclear power plants and 12 occurred at other licensees (e.g., fuel cycle facilities, industrial radiographers, medical institutions). The details of the 19 AOs at NRC licensees were also published in the Federal Register. Five of the 19 events reported in 1986 occurred during CY 86, 13 occurred during CY 85, and one occurred late in CY 84.

The second quarter CY 86 AO report was approved by the Commission in January 1986 and issued in February 1986. The report contained seven AOs at NRC licensees (two AOs occurred at the nuclear power plants and five occurred at other NRC licensees), and two AOs at Agreement State licensees. The report also described seven additional events as other reports of interest.

The third quarter CY 86 AO report was sent to the Commission for approval by SECY-87-73 dated March 18, 1987. The proposed report contains five AOs at NRC licensees (four of the AOs occurred at nuclear power plants), and one AO at an Agreement State licensee. The report also describes four additional events as other reports of interest.

The fourth quarter CY 86 AO report is presently under development by the staff. It is expected that it will propose nine AOs at NRC licensees (three of the nine AOs are associated with nuclear power plants) and no AOs at the Agreement State licensees. The report is also expected to describe four additional events as other reports of interest.

A summary of the 1986 AOs are provided in Appendix A. These include those approved by the Commission (as of late February 1986) and those still in the approval chain.

5.2.4 Issue Operational Data Feedback Reports, Including Bimonthly Power Reactor Events and Monthly LER Compilation Reports

In 1986, AEOD continued with the routine publication of operating experience feedback. Two routine publications issued are discussed in this section.

Power Reactor Events is a bimonthly newsletter intended to feed back operating experience information and lessons learned to personnel at commercial nuclear power plants. These personnel include licensing engineers, plant managers, and training coordinators, as well as reactor operators and support personnel. Other recipients total approximately 1000 and include NRC employees; employees of DOE and other Federal agencies; applicable State agencies; various domestic and foreign industry groups; and interested individuals.

Each issue contains the following information:

- ° "Summaries of Events" provides detailed write-ups of events that may be significant because of their safety implications and/or because of operator and licensee action taken during and after the events.

- ° "Excerpts of Selected Licensee Event Reports," added in June 1984, provides direct excerpts from LERs. In general, the information describes conditions or events that are unusual or complex, or that demonstrate a problem or condition that may not be obvious.
- ° "Abstracts of Other NRC Operating Experience Documents" provides abstracts and/or titles from pertinent NRC documents reflecting operational experience such as Abnormal Occurrence reports, IE Bulletins and Information Notices, AEOD case studies and engineering evaluations, NRR generic letters, and NRR operating reactor event memoranda.

In CY 86, six issues of Power Reactor Events were published. They included 52 event summaries and 66 LER excerpts. Consideration is being given to terminating this publication, since the study of licensee programs mentioned in Section 3.1.1 indicated the document, while read and used by some licensees as source material for training programs, was not resulting in widespread use by licensees to initiate voluntary actions in response to the lessons communicated. The decision to terminate is pending the results of three inputs--completion of a contract initiated by IE to improve generic communications with licensees, a survey of users planned for spring 1987, and possible impacts from the upcoming major reorganization of NRC functions.

Twelve issues of the monthly LER Compilation were published using data from the SCSS/RECON databases. This compilation provides an abstract of each event sorted by facility name and chronologically by event date. Each event is also cross-referenced by system, component, component vendor, and general keyword indexes.

5.2.5 Prepare Reports on U.S. Events to the Nuclear Energy Agency's Incident Reporting System (NEA-IRS), Provide Support to the International Atomic Energy Agency's Incident Reporting System (IAEA-IRS), and Support International Data Usage Programs

The U.S. continued to participate during the reporting period with 12 other countries (Belgium, Canada, Finland, France, Federal Republic of Germany, Italy, Japan, Netherlands, Spain, Sweden, Switzerland, and United Kingdom) in a centralized Incident Reporting System (IRS) for exchanging information on operational experience. This system is operated by the Nuclear Energy Agency (NEA) under the direction of the International Committee on the Safety of Nuclear Installations (CSNI).

AEOD serves as the principal U.S. technical representative to Principal Working Group (PWG) No. 1, "Operating Experience and Human Factors." During this period, PWG-1 met in September 1986 with representatives from IAEA countries to discuss significant operating events and also met in September 1986 to evaluate the significance of operating events and data in NEA countries. A total of four technical presentations were made by the AEOD representative in these meetings. In addition, AEOD provided suggestions to both organizations regarding ways to obtain better consistency in the quality and the quantity of reporting.

AEOD screens U.S. operating experience to select and prepare reports on those events meeting pre-established reporting criteria. During this report period, AEOD prepared and submitted to NEA a total of 65 IRS reports on U.S. operational events. In addition, AEOD prepared and submitted to NEA supplemental information on several U.S. events that were of particular interest to other IRS participants.

In 1985, the U.S. formalized its commitment to support the IAEA-IRS. The U.S. is fulfilling its commitment through the NEA-IRS with NEA coordinating with IAEA on the exchange of incident reports. AEOD, as mentioned above, also provided support to IAEA in 1986 on significant operating events and in the program direction area.

AEOD also provided considerable support to the NEA's International Symposium on Reducing Reactor Scram Frequency, held in Japan in April 1986. The objectives of the symposium were to: (1) compile consistent statistical data regarding scram frequencies in NEA member countries, (2) exchange insights on the reasons for the differences between countries, and (3) look for measures by which plant performance can be improved. AEOD served on the program committee, coordinated the U.S. participation, and presented two papers and chaired three sessions at the meeting. One of the papers was a summary overview of the responses at the member countries to an NEA questionnaire on their scram experience. The other was the Trends and Pattern Analysis of Unplanned Reactor Trips at the U.S. Light Water Reactors in 1984 and 1985 (AEOD/P504).

In addition, subsequent to the Chernobyl accident, the Subcommittee on Energy, Nuclear Proliferation, and Government Process of the Committee on Governmental Affairs, U.S. Senate, requested that the NRC participate in a hearing on international nuclear safety concerns. AEOD provided technical support to the Office of International Programs, attended the hearing on May 8, 1986, and provided responses to Congressional questions for the record subsequent to the hearing (S. HRG 99-669).

5.2.6 Develop Techniques to Apply PRA Perspectives to the Screening and Analysis of Operational Events

An AEOD goal is to use the results and methods of probabilistic risk assessment (PRA) in reactor operational event assessment. Because of the large number of LERs received, routine use of PRA for prioritizing LERs may not be practical. Yet such techniques may offer promise to help assess and "calibrate" the significance of selected operating events and component failures.

In 1985, AEOD assumed responsibility for the Accident Sequence Precursor (ASP) Program from the Office of Nuclear Regulatory Research. The objective of this program is to systematically evaluate operational data from U.S. nuclear power plants. The evaluation is intended to assess potentially serious operational incidents from a wide perspective. The information from this program is intended to serve as one element in NRC's assessment of nuclear plant operational safety.

The ASP program owes its genesis to the Risk Assessment Review Group, which concluded that "unidentified event sequences significant to risk might contribute...a small increment...[to the overall risk]" (NUREG/CR-0400). The report continues, "It is important, in our view, that potentially significant (accident) sequences, and precursors, as they occur, be subjected to the kind of analysis contained in WASH-1400." Evaluations done for the 1969 to 1981 period [NUREG/CR-2497 (1969-79) and NUREG/CR-3591 (1980-81)] were the first efforts in this type of analysis.

The ASP program involves a systematic review of operational events that have occurred at light-water power reactors to identify and categorize precursors to potential severe core damage accidents. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition in which core cooling may not be adequate.

5.2.6.1 ASP Evaluation of Reactor Operational Data

A nuclear plant is designed with safety systems for mitigating accidents or off-normal initiating events that may occur during plant operation. These systems have a finite probability of failing or being in a failed state when required to operate. The ASP program uses LERs and other plant data, estimated system unavailabilities, the expected average frequency of initiating events [loss of feedwater, loss of offsite power, loss-of-coolant accidents (LOCAs)], and event details to evaluate the potential impact of the following two situations:

- (1) Safety System Unavailability. Given a failure of a safety system or partial failures in two or more systems, the expected initiating event occurrence rates are used to determine the number of initiating events that may challenge the failed and backup systems during the period associated with the failure. The expected challenges are multiplied by system failure probabilities, using event trees to evaluate the likelihood of the occurrence of the overall event sequence.
- (2) Initiating Event Occurrences. The probability exists that the standby safety systems may fail when called on to mitigate expected transients or transient-initiating events. The likelihood of potential severe core damage for precursors that included initiating events is calculated based on expected response of the safety systems. Failed or degraded systems existing at the time of the initiating event are accounted for in the calculations.

Initiating event frequency and system failure probability estimates are used, in conjunction with precursor event trees, to estimate a conditional probability of potential severe core damage associated with each precursor. This probability is an estimate of the chance of potential severe core damage (inadequate core cooling), given that the precursor event occurred in the manner it did, and can be considered a measure of the residual protection against potential severe core damage available during the event. The conditional probabilities associated with each precursor are used to evaluate precursors as to significance and to identify dominant sequences among all postulated sequences to potential severe core damage for the more significant events.

5.2.6.2 Recent AEOD Publications

In 1986, AEOD published two reports: "Precursors To Potential Severe Core Damage Accidents: 1985 - A Status Report" (NUREG/CR-4674), and "Precursors to Potential Severe Core Damage Accidents: 1984 - A Status Report" (Draft). (Following the transfer of ASP responsibilities to AEOD, the 1985 report was performed before the 1984 report in order to develop an understanding of the most current data.)

Studies of this nature are subject to certain inherent limitations. Often there is limited data, and the results may be affected by many of the decisions inherent in the process as well as in the methodology itself. However, a determined effort is made to address these problems. Although uncertainties exist in the numeric probability estimates associated with each event addressed in the report, the identification of the more serious events from a core damage standpoint is considered reasonably valid. The significant findings of these two studies are highlighted here.

Accident sequences of interest in these studies are those that, if completed, would have resulted in inadequate core cooling in the short term (typically up to 20 to 30 minutes) and then would have potentially resulted in severe core damage. Accident sequence precursors are events that are important elements in such accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition leading to severe core damage. Precursors were selected and evaluated using a similar screening process and significance quantification methodology as was used in NUREG/CR-3591 for the 1980 to 1981 period. This methodology permits a reasonable quantification of the significance of an event without the laborious detail associated with evaluation using event trees and fault trees down to the component level, while including observed human and system interactions.

The 1985 Study

Approximately 3000 LERS concerning events that occurred during 1985 were screened for accident sequence precursors. Of these, a total of more than 1400 LERS were selected for detailed review if they included a reactor trip or more serious initiator, included two or more component failures or unavailabilities, or described an event that proceeded differently than expected. All LERS selected for detailed review were subjected to an indepth evaluation. As a result of this detailed review, 63 events were selected as precursors. As a result, certain preliminary conclusions can be drawn.

The number of auxiliary feedwater (AFW), reactor core isolation cooling (RCIC), and isolation condenser failures following an initiating event observed in 1985 was greater than expected. In the case of AFW, three recoverable failures on demand following an initiating event were seen. No AFW system failures during testing were reported. Such failures would be expected more frequently than those following initiating events, because testing occurs more frequently, if testing adequately addressed potential failure modes.

Although the number of precursors per reactor year identified in 1985 increased over the number identified in earlier reports (primarily reflecting the revised LER requirements and changed program emphasis), the number of more significant events (i.e., with higher conditional probabilities of potential severe core damage) appears about the same as in the 1980 to 1981 time period. This observation is based on analyses that used system failure probabilities consistent with those used in the 1980 to 1981 evaluations. The use of higher failure probabilities for AFW, RCIC, and isolation condensers, which may be indicated by the number of failures seen in these systems in 1985, has not been factored into the precursor conditional probability estimates.

The dominant severe core-damage accident sequences associated with the more important precursors for BWRs were generally consistent with those predicted in past probabilistic risk assessments (PRAs) and previously observed in this program for BWRs. For PWRs, the number of small-break-LOCA-related events previously predicted (and seen) was not observed. This resulted in a relatively greater emphasis for sequences associated with loss of all secondary cooling and failure of feed and bleed for PWRs.

The 1984 Study

Approximately 2400 LERs concerning events that occurred during 1984 were screened for accident sequence precursors. Of these, a total of more than 900 LERs were selected for detailed review if they included a reactor trip or more serious initiators, included two or more component failures or unavailabilities, or described an event that proceeded differently than expected. Forty-eight operational events reported in LERs during 1984 are considered to be precursors to potential severe core damage.

The number of precursors per reactor year identified in 1984 increased over the number identified in 1980-81 to the same level previously noted for 1985 events (about 0.6/reactor year). Part of this increase is believed to reflect the changed program emphasis and revised LER reporting requirements (NUREG-1022). In addition, the frequency of events with conditional probabilities of core damage greater than 1×10^{-4} was twice that observed in 1980-81. However, many of these events were only slightly above 1×10^{-4} , and only one event greater than 1×10^{-3} was observed, compared to six in the 1980-81 period.

The dominant severe core damage accident sequences associated with the more important precursors for BWRs were generally consistent with those predicted in past probabilistic risk assessments (PRAs) and previously observed in this program for BWRs. For PWRs, the number of small-break LOCA-related events previously predicted (and seen) was not observed. As was the case with the 1985 data, this resulted in a relatively greater emphasis in PWRs for sequences associated with loss of all secondary cooling and failure of feed and bleed.

5.2.6.3 Future Plans

The ASP evaluations of LERs occurring through 1986 will use the "potential for core damage" as the primary measure of LER significance. LERs reported in 1987 and later will be evaluated and documented in ASP NUREG reports using the Commission Safety Goals as the measure of event significance, to the extent practical. This means that the program goal will be to measure the operational event significance

to consider the likelihood and extent of radiological release through containment to the environment. In order to perform operational data evaluations using the Commission Safety Goals, new methods beyond those currently used for estimating "core damage likelihood" are needed. Models and program results from the Office of Research's "Reactor Risk Reference Report" (NUREG-1150) are planned to be used to augment ASP assessment of LERs using the Commission Safety Goal measures of significance.

5.2.7 Develop and Manage Data Bases

AEOD utilizes a contract with the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory to assist in the data storage, retrieval, and evaluation efforts, as discussed below:

(1) Sequence Coding and Search System (SCSS)

The SCSS was developed with two objectives: (1) encode all of the relevant technical information provided by the licensee in the LER, and (2) encode the information with sufficient "tags" so that the individual pieces can be precisely retrieved. The SCSS data base currently contains data from 1980 to the present.

- ° During this period, approximately 3300 current LERs were added to the data base. Also, due to the backfit program, approximately 3100 LERs from 1980 were added. This increases the number of LERs on the data base to almost 24,280.
- ° Amendments 4 and 5 to Revision 1 of the controlled copies of the SCSS Coder's Manual were issued in January and August 1986, respectively.
- ° Revision 1 of the Quality Assurance Program for SCSS was issued in April 1986.
- ° An amendment to the SCSS Programmer's Manual reflecting additions and changes to the software and procedures which were made during FY 86 was issued in September 1986.
- ° In January 1986, an IBM disk dedicated to support SCSS computer activities was installed to accommodate the 20,000+ LER records and other supplemental files.
- ° In March 1986, the coding process was modified to begin coding LERs directly on-line interactively versus coding on paper forms and submitting these forms for data entry via keypunch operations. The QA review and scan operations were performed on-line as well. The computer QA checks are executed at each of the three processing stages, thus providing immediate feedback at each stage of the overall process.
- ° By March 1986, the majority of LER abstracts were being entered via an optical character reader (OCR) for subsequent matching with the SCSS matrix.

- ° Plant specific data was added to the SCSS data base. This information included design electrical rating, criticality date, commercial operation date, the State in which the plant is located, and the number of coolant loops.
- ° Work was completed in September 1986 to backfit the SCSS data base to include LER data from 1980.
- ° There were 11 requests for on-line access to the SCSS during 1986 from employees of the NRC and the DOE national laboratories. This made a total of 53 individuals who have direct access to the data base.
- ° AEOD and its contractor responded to about 215 requests for data output from SCSS from individuals without direct access.

(2) Foreign Event File (FEF)

Beginning in 1980, NOAC developed the Foreign Event File (FEF) data base, and in late 1985 placed it (as a protected file) on the DOE computerized information system (RECON) to increase its availability to authorized users. On December 31, 1986, the RECON system was terminated and the FEF was transferred to DOE's Integrated Technical Information System (ITIS). The FEF file is similar to the abstract file of LERs that has also been maintained on RECON by the Nuclear Safety Information Center (NSIC), which is part of NOAC. The scope of the Foreign Event File is limited to events at foreign light water reactors (LWRs) with an output greater than 200 MWe.

- ° During the period, 903 events were added to the FEF data base and events identified as potentially safety significant and possibly applicable to U.S. reactors were forwarded for further review to AEOD, NRR, RES, IE, and INPO. FEF currently contains information on about 7000 foreign events (1120 with full abstracts).
- ° During 1986, the principal users of the system were NOAC and the NRC. The file was accessed 205 times.
- ° In 1986, AEOD requested that NOAC review foreign reporting and suggest methods for obtaining consistency in quality and quantity in both the NEA-IRS and IAEA-IRS. AEOD then communicated its recommendations to the NEA and IAEA for consideration.

(3) RECON LER file

An LER abstract and search capability is also maintained on the DOE RECON system, a computerized information retrieval system designed to provide users with remote terminal access to bibliographic data bases. RECON access is available on-line to DOE offices, to institutions holding DOE contracts, to other Federal agencies with energy-related or energy intensive missions, and to State agencies with State-wide responsibilities for energy programs or information. Printouts from the NOAC data base are also available to other organizations on a cost-recovery basis by request. From 1963 to 1983, the LER data for this data base was manually

abstracted, and keywords were manually assigned. Once the SCSS became operational, the NSIC data was generated by the computer from the SCSS data base. Therefore, with minimal resources, the NSIC LER file continued to be available on-line to the government agencies and contractors. However, in 1986, AEOD was informed that RECON would be terminated as of 12/31/86. Therefore, RM and AEOD have worked to set up the NSIC data base on a commercial data base network or another DOE system. Negotiations are in progress.

- ° Over 4400 LERs were added to the RECON database which now contains approximately 50,700 LERs received since 1963.
- ° Of the more than 50 data bases included in the RECON system, the LER data base is typically in the top six with respect to the frequency of use. For example, for the last 3 months of 1986, the number of accessions to this data base ranged from 36 to 68 per month.
- ° AEOD and its contractor responded to about 40 requests for data output from the RECON LER file.

(4) Access INPO Data Files

Included in the Memorandum of Agreement with INPO is a provision to share data bases and results of analyses. As a result, AEOD maintains the capability to conduct searches of selected INPO data files such as NPRDS.

During this period, AEOD, along with RM and the EDO, worked with INPO to resolve problems concerning the need for NPRDS output. These problems resulted from the need to provide access for additional NRC users, to train users on the on-line NPRDS data base management system SEEK, and to perform NPRDS searches for the NRC. Thus the number of individuals outside of AEOD who were authorized to have access remained at seven. Late in 1986, contract negotiations were begun in response to INPO's request for cost-recovery of the NRC's use of the NPRDS data. It is anticipated that the contract will be finalized early in 1987.

AEOD responded to about 45 requests for data output from NPRDS.

(5) Formulate and Operate Internal Management Information Systems

AEOD maintains several internal data bases for such purposes as documentation of screening results and resource accountability. An additional small data base is maintained on reports of U.S. events to NEA/IRS.

Routine reports of LER screening results (WAMS) and manpower utilization (TACS) are prepared periodically and used by AEOD managers.

(6) Other Activities

PTB prepared or coordinated office procedures and prepared the office's input to the NRC Annual Report to Congress. Program support reports completed by PTB are listed in Table 15.

Table 15

Program Support Reports

<u>Date</u>	<u>Subject</u>	<u>No.</u>
1/86	Trends and Patterns Program Plan - FY86-FY 88	P601
5/86	An Overview of Nuclear Power Plant Operating Experience Feedback Programs	S602
8/86	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1985	P602
8/86	Trends and Patterns Analysis of Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Plants (July 1984-December 1984)	P603
8/86	Trends and Patterns Analysis of the Operational Experience of Newly Licensed U.S. Nuclear Power Reactors	P604

5.3 Nonreactor Assessment Staff (NAS)

NAS screens and analyzes the operating experience associated with the activities and facilities licensed by the Office of Nuclear Material Safety and Safeguards (NMSS) and by Agreement States (i.e., nonreactor operating experience). In addition, NAS conducts studies from a human factors perspective on both reactor and nonreactor operating events.

The major activities of NAS include: reviewing events reported to the five Regional Offices by licensees; reviewing inspection reports; entering coded information on events identified from the review into computer data files; evaluating the events as a whole; and performing detailed reviews of specific events and concerns. The full scope of NAS responsibilities is shown below:

NONREACTOR ASSESSMENT STAFF KATHLEEN BLACK, CHIEF

Responsibilities and Work Products

- 5.3.1 Screen individual events associated with NMSS and Agreement State licensed activities and facilities and determine significance
- 5.3.2 Analyze and evaluate individual nonreactor and medical misadministration events and related potentially generic safety concerns
- 5.3.3 Analyze reactor and nonreactor events from a human factors perspective

5.3.4 Document independent technical assessments in:

- Case studies
- Engineering Evaluations
- Technical Reviews
- Memoranda

5.3.5 Develop, maintain, and provide updating data to these computerized data files:

- Nonreactor data file
- Medical misadministration data file

The milestones and progress on these NAS responsibilities are summarized individually below.

5.3.1 Screen Individual Events Associated with NMSS and Agreement State Licensed Activities and Facilities and Determine Significance

About 6000 reports per year are received from the approximately 8300 NRC licensees. Of these reports, approximately 400 involved an operational event, such as an overexposure, spill, or medical misadministration. In addition, the 8000 Agreement State licensees are required to submit reports to the cognizant Agreement State which, in turn, provides reports on certain of the reported events to the NRC. The Agreement State reports are also reviewed principally from the standpoint of generic problem identification and from their contribution to the review of events reported by NRC licensees.

As a result of these NAS screening activities, the following individual events associated with NMSS and Agreement State licensed activities and facilities were identified: (1) events that are determined to be AOs; (2) events that are themselves not of high individual significance but are events with potential generic significance; and (3) events that appear to have an immediate significance, either specifically or generically.

- Of the many hundreds of event reports from NRC and Agreement State licensees, 16 were determined by the Commission to meet the AO reporting criteria. Another eight events occurring in 1986 were recommended by AEOD as meeting the criteria, but were not yet acted on by the Commission. Of the 24 events, 14 occurred in 1986 and ten occurred in 1984 or 1985; four were diagnostic misadministrations, five were therapeutic misadministrations, and four were occupational overexposures (three of which were overexposures of radiographers or assistant radiographers). There were 11 additional AOs: exposure of a member of the public; the rupture of a UF-6 cylinder at Sequoyah Fuels; loss of management controls at an irradiator licensee; contamination of a scrap steel facility; two willful failures to report misadministrations; a release of krypton-85; distribution of molybdenum-contaminated technetium; contamination of a building at Wright-Patterson Air Force; suspension of the licensee for a radiography and teletherapy servicing firm; and removal of an officer of a radiography firm from any assignment of position related to NRC licensed activities. In one AO, a member of the public was exposed.

Each of the four diagnostic misadministrations involved the misadministration of iodine-131.

The five therapy misadministrations all resulted from human factors. AEOD published a case study (AEOD/C505) on therapy misadministrations that had occurred prior to 1985. A major conclusion of this case study--that independent verification of dose calculations and patient chart reviews could reduce the number of therapy misadministrations--is further substantiated by these AOs.

5.3.2 Analyze and Evaluate Individual Nonreactor and Medical Misadministration Events and Related Potentially Generic Safety Concerns

During the reporting period, case study AEOD/C601, "Rupture of an Iodine-125 Brachytherapy Source at the University of Cincinnati Medical Center," was completed and issued August 12, 1986. The study documents the analyses of the rupture of an iodine-125 seed (nominal activity of 40 millicuries) used in brachytherapy treatment of brain tumors at the University of Cincinnati in September 1984.

The primary conclusion of the study was that the risk of an iodine-125 seed rupture is relatively high when the seeds are reused for several patients. The risk of a seed rupture is associated with:

- ° The susceptibility of the seeds to damage from typical tools used for removing the seeds (razor blade, scissors, etc.); and
- ° The discolored or stained condition of the catheters after use in therapy, making viewing of the seeds difficult.

The consequence of the seed rupture at the University of Cincinnati, involving patient and other personnel uptakes and facility contamination, could have been mitigated by adequate radiation surveys of the work area and the tools used to remove the seeds from the catheter, or by performing a leak test of the seeds. Additionally, personnel uptakes other than the patient and the facility contamination could have likely been prevented if the seed removal operation had been performed under a fume hood.

It appears that the consequence (personnel uptakes, and personnel and facility contamination) of a similar event could also be mitigated by employing radiation safety procedures designed to promptly detect if a seed is ruptured and to prevent personnel uptakes and personnel and facility contamination. Such procedures would include: performing the removal/reloading operation in a fume hood; performing wipe surveys of tools and the area used for the removal and reloading of the seeds; or leak testing the seeds following the removal/reloading operation.

The report recommended that the Office of Inspection and Enforcement send an information notice to the affected licensees describing the event and the action taken by the licensee and the source manufacturer to prevent the recurrence of similar events; specific instructions and safety precautions for reusing the seeds be communicated; and NMSS should explore the option of addressing the reuse of the high activity iodine-125 seeds during the license issue, renewal, or amendment process.

5.3.3 Analyze Reactor and Nonreactor Events from a Human Factors Perspective

- ° In December 1985, AEOD issued case study report AEOD/C504, "Loss of Safety System Function Events." The study identified 133 events involving a loss of safety system function (LSSF) in the 1981 to June 1984 period, 87 of which (65%) were the result of human factors contributions. The results of the study were a principal impact and were published as part of a November 1986 study by the CSNI Principal Working Group #1 (Operating Experience and Human Factors) report entitled "Loss of Safety System Functions."
- ° Special study AEOD/S401, "Human Error in Events Involving Wrong Unit or Wrong Train," was published in January 1984. Because of the implications of wrong unit/wrong train events, NAS has continued to track and analyze similar subsequent events including events involving the wrong component.

Because wrong unit/wrong train/wrong component (WU/WT/WC) events continued to occur, AEOD and the Office of Nuclear Reactor Regulation (NRR) undertook a cooperative field survey of the problem during 1985. A series of site visits was undertaken to obtain first-hand knowledge of plant layout, equipment identification, and personnel practices. By the end of 1985, NRR and AEOD staff had completed site visits and reviews at ten multi-unit sites to obtain information on the factors that appeared to have contributed to these types of events. NRR published the results of the study in "An Investigation of the Contributions to Wrong Unit or Wrong Train Events," NUREG-1192, dated April 1986.

Industry has recognized the potential significance of this type of event, and NUMARC has stated that it will study this issue. At the suggestion of NRR, AEOD will continue to monitor the occurrence of this type of event on a normalized basis in order to determine the progress made by industry. Reports on results of the monitoring will be published semiannually.

Technical review AEOD/T607, dated September 19, 1986, includes the annual total number of WU/WT/WC events over the 1981 to June 1986 time period. The report concluded that: (1) there does not appear to be a reduction in the number of events involving the wrong unit/wrong train/wrong component, and (2) plants with little operating experience (less than 2 years from initial criticality) seem to experience a disproportionately high number of these events. Preliminary data for the second half of 1986 confirm this (see Table 16). However, the apparent high rate of occurrence in more recent years may reflect the improved investigation and reporting of the details of causes of events, rather than an increase in event occurrence.

- ° Of the significant operating reactor events recently investigated by the NRC, a large fraction involved procedures to some degree. As a result, AEOD began a study of 1984 and 1985 events that involved procedures and that were significant from a safety standpoint. The initial study focused on 1984 and 1985 LERs that were assigned an AEOD significance level of "1" or "2". Of the 179 such reports from 1984 and the 110 reports from 1985, totals of 54 and 47 reports, respectively, were found to

involve procedures.* The 54 events in 1984 and the 47 events in 1985 that involved procedures were then examined to determine the characteristics of the events.

The initial review of the characteristics of the events indicated that problems associated with procedures were contributing factors in a high percentage of recent significant events, and that some licensees are more prone to these problems than others. However, the data also indicated that some plants may tend to use procedure revisions as a general corrective action following an event. This may be due to a lack of understanding as to the fundamental cause and contributing factor (perhaps due to lack of thorough event investigation); (2) a focus on equipment problems rather than the personnel performance problems due to deficiencies in personnel qualifications, training, or environment; and/or (3) a desire to quickly "close out" an LER, when the complete action to prevent recurrence is far-reaching.

To achieve a better understanding of the contribution of procedures to the occurrence of significant events, a series of site visits and discussions with resident inspectors and licensees' staff was undertaken beginning in late 1986. The site visits should be completed and the special study report issued in early 1987.

Some preliminary conclusions of the study are that problems associated with procedures were contributing factors in a high percentage of recent (1984-1985) significant events; and licensees frequently reported that procedures were incomplete, possibly indicating inadequate programs for assuring procedure quality (e.g., procedure improvement programs).

5.3.4 Document Independent Technical Assessments

Technical Reviews:

- ° AEOD/T604, "Events Resulting from Deficiencies in Labeling and Identification System," dated May 6, 1986, described the characteristics of 38 recent events that resulted from deficiencies in nuclear power plant labeling and identification (L&ID) systems; that is, administrative systems for ensuring the correct identification of units, trains, and components. The report provides additional evidence that deficiencies in L&ID systems frequently contribute to human error and that this continues to be a problem in the nuclear power industry.

*For the purpose of the review, "events resulting from defective procedures" were those events that were at least partially the result of a lack of, deviation from, or deficiencies in, operating, maintenance or administrative control procedures. An event was deemed to be the result of a defective procedure if (1) the LER stated that a procedural deficiency or deviation from a procedure was a contributing factor or that a procedure change(s) had or would be made to prevent recurrence, or (2) in LERs in which "procedures" were not explicitly mentioned, a judgement was made as to whether procedures were a contributing cause.

Table 16

Events Involving the Wrong Unit, Train, or System Considering
Event Type and Plant Operating Experience*

	<u>1981</u>	<u>1982</u>	<u>1983</u>	<u>1984</u>	<u>1985</u>	<u>1986</u>	<u>Total</u>
Wrong Unit Events at Plants with:							
>2 years experience	2	3	4	3	8	5	25
≤2 years experience	1	0	1	3	1	3	9
Wrong Train Events at Plants with:							
>2 years experience	6	8	12	18	14	25	83
≤2 years experience	1	1	1	4	8	5	20
Wrong Component Events at Plants with:							
>2 years experience	N/A**	N/A**	4	10	19	33	66
≤2 years experience	N/A**	N/A**	0	7	22	18	47
Total Events at Plants with:							
>2 years experience	8	11	20	31	41	63	174
≤2 years experience	2	1	2	14	31	26	76
Total	10	12	22	45	72	89	250
Number of Multi-Unit Sites	20	20	23	25	26	28	
Total Number of Operating Reactor Plants***	69	74	77	83	92	99	
Number of plants with >2 years experience	63	66	69	74	77	83	
Number of plants with ≤2 years experience	6	8	8	9	15	16	

*Experience is defined as the amount of time from initial criticality.

**N/A = Not Available

***Totals do not include LaCrosse and Fort St. Vrain.

Total does include TMI-1 for 1985 and 1986.

- ° AEOD/T607, "Occurrence of Events Involving Wrong Unit/Wrong Train/Wrong Component - Update through June 1986," dated September 19, 1986.

Other Reports:

- ° "Wrong Unit/Wrong Train Events, 1981-1985," dated February 13, 1986.
- ° "Trip Report for McGuire and Oconee Site Visits Regarding Wrong Unit/Wrong Train Events," dated March 6, 1986.
- ° "Trip Report for Turkey Point Site Visit Regarding Wrong Unit/Wrong Train Events," dated March 6, 1986.

5.3.5 Develop, Maintain, and Provide Updating Data to the Nonreactor and Medical Misadministration Data Files

Computerized data files continued to be updated by NAS on (1) nonreactor events, and (2) medical misadministrations.

From the events screened by NAS, approximately 200 nonreactor events and 400 misadministrations were coded by NAS and entered into the nonreactor data file during 1986.

5.4 Incident Investigation Staff (IIS)

The overall goal of the Incident Investigation Program (IIP) is to promote the public health and safety by thoroughly understanding the nature and causes of significant events and thereby reduce the frequency and consequences of such incidents. The objective of the IIP is to ensure that operational events are investigated in a systematic and technically sound manner to gather information pertaining to the probable causes of the events, including any NRC contributions or lapses, and to provide appropriate feedback regarding the lessons of experience to the NRC, industry, and public. By focusing on probable causes of operating events and identification of associated corrective actions, the results of the IIP process should improve nuclear safety by ensuring a complete technical and regulatory understanding of significant events.

The IIP has two investigatory responses based on the safety significance of the operational events. Both involve responses by an NRC team to determine the circumstances and causes of an operational event. For the most significant operational events, an Incident Investigation Team (IIT) is dispatched by the Executive Director for Operations (EDO) to investigate the event. The response to less significant operational events are designated Augmented Inspection Teams (AITs) which are usually initiated and directed by the Regional Office. Although no IITs were activated during CY 1986, nine AITs were conducted, as shown in Table 17.

These AIT and IIT responses ensure that significant operational events are investigated in a manner that is timely, objective, systematic, and technically sound; that factual information pertaining to the event is documented; that probable cause(s) are ascertained; and that a complete technical and regulatory understanding of the event is achieved.

Table 17

Augmented Inspection Team (AIT) Responses

<u>Date of Event</u>	<u>Facility</u>	<u>Event/Comments</u>
1/4/86	Sequoyah Fuels Facility	Rupture of Model 48Y UF-6 Cylinder and Release of Uranium Hexafluoride
1/31/86	Perry	Earthquake - 5.0 on Richter Scale
4/12/86	Pilgrim	Recurring Residual Heat Removal (RHR) Isolation and Valve Leakage
5/19/86	Palisades	Multiple Equipment Failures, Maintenance Backlog
6/1/86	LaSalle Unit 2	Feedwater Transient with Failure of Static-O-Ring Switches
6/27/86	Catawba Unit 2	Unit 2 Depressurization During Remote Shutdown Test
9/11/86 9/19/86	Hope Creek	Multiple Failures During Loss of Offsite Power Tests Faulty Bailey Solid State Logic Modules
12/3/86	Hatch	Loss of Water from Spent Fuel Pools to Site Due to Failed Inflatable Seals
12/9/86	Surry Unit 2	Main Feed Pump Suction Pipe Rupture

In January 1985, the Incident Investigation Staff (IIS) was established as an organizational element within AEOD. The IIS has the responsibility to develop policy, program requirements, and procedures for IIT investigations of significant operating events. An IIS member will accompany each IIT to provide administrative support, liaison, and technical guidance to ensure that the IIT activities are consistent with established procedures and coordinated with NRC offices and other organizations. The IIS maintains and integrates plans, procedures, team rosters, and training, and coordinates staff activities to achieve the IIP objectives as noted below:

INCIDENT INVESTIGATION STAFF
WAYNE D. LANNING, CHIEF

Responsibilities and Work Products

- 5.4.1 Develop formal guidance for and issue the NRC Incident Investigation Program (Manual Chapter 0513)
- 5.4.2 Prepare and maintain personnel rosters of candidate IIT leaders and members
- 5.4.3 Develop and provide training for IIT candidates
- 5.4.4 Develop formal protocol for the treatment of guaranteed equipment and information during an IIT
- 5.4.5 Provide and maintain administrative support for the IITs

The following describes the activities associated with the IIP that were completed during CY 1986.

5.4.1 Issuance of Manual Chapter NRC-0513, "NRC Incident Investigation Program"

An NRC Manual Chapter was prepared that defines the duties, responsibilities, authorities, and schedule for event investigation under the IIP. This Manual Chapter contains guidance regarding the significant operational events to be investigated by an IIT and by an AIT. The Manual Chapter was issued for comment on March 5, 1986, concurred by all offices, forwarded to the EDO for approval on July 31, 1986, and issued final on August 8, 1986.

5.4.2 Establishment of Personnel Rosters

Personnel rosters were developed to enable a timely identification of NRC staff having specific expertise to be considered as IIT leaders and members for events at reactor sites. Candidates for the roster were recommended by their office director and approved by the EDO. The rosters are updated and revised every 6 months.

The initial roster was completed in February 1986 and approved by the EDO in April 1986. In August, the rosters were revised to update each member's expertise based on the background information provided. The rosters and

background information have been entered into a database file using an IBM personal computer for ready access and convenience for updating and revising.

5.4.3. Development of an IIT Training Program

The purpose of the IIT training program is to provide IIT candidates with a comprehensive range of guidelines and methodologies to ensure that NRC investigations of significant events are timely, thorough, coordinated, and formally administered. The training program was developed and organized by the IIS after discussions with representatives from the National Transportation Safety Board, the Federal Aviation Administration, Ontario Hydro, the Department of Energy, and the National Aeronautics and Space Administration. The experience and insights from previous IIT team leaders and members were incorporated into the training course.

The course consists of an intensive 2-week curriculum that begins with an overview of the NRC Incident Investigation Program, including perspectives from previous IITs, and covers the following areas:

- Incident Investigation Program
- Investigation Perspectives
- Investigation Guidelines
- Investigation Analytical Techniques

The two-week training course covering event investigation methodologies and the IIT procedures was given to 25 representatives from Headquarters and Regional Offices on July 7-18, 1986. This class composition was selected on the basis of establishing five IITs. As part of the training course, a training manual was prepared for IIT class participants containing background material and draft procedures.

5.4.4 Development of Standard Confirmation of Action Letter and Order

A generic Confirmation of Action Letter (CAL) and Order were developed that establish the protocol for the treatment of quarantined equipment and information during an IIT investigation. The licensee's responsibility with respect to the quarantined equipment is clearly defined. The generic CAL and Order also address parallel investigations by confirming that the licensee will ensure that any investigation to be conducted by the licensee or third party will not interfere with the IIT investigation. These documents have been included in the Incident Investigation Manual.

5.4.5 Development and Implementation of IIT Administrative Requirements

The IIS developed procedures to guide the administrative functions necessary to dispatch, support, and close out IITs. The completion of these administrative procedures was a major activity during CY 1986. The administrative procedures address the following subject:

5.4.5.1 Imprest Fund

An imprest fund was established in AEOD to enable IIT members to obtain a travel advance if an IIT is activated during off-duty hours for IIS and IIT

travel during non-duty hours. The IIS has obtained a safe and cash box, and has implemented appropriate administrative controls for safeguarding and dispersing this fund.

5.4.5.2 Travel Arrangements

- ° Ability to process travel authorizations and obtain travel advances on short notice regardless of day or time. Procedures address the use of transportation requests and accessing the imprest fund for emergency travel.
- ° Ability to identify the proximity of the major airports and available lodging near sites. The IIS maintains a Site Data Notebook containing information enabling IITs to travel to the site, with confirmed hotel reservations (e.g., proximity of major airports, available lodging near sites, reactor site addresses, and directions to site).
- ° Ability to process travel claims in a timely manner. Procedures have been established to ensure timely and correct submission of travel vouchers for IIT members.

5.4.5.3 Stenographers

The IIS is responsible for ensuring that stenographers are available at the site in a timely manner to transcribe IIT interviews. The Atomic Safety and Licensing Board Panel (ASLBP) manages the NRC contract for stenographers and assists the IIS in obtaining this service. Procedures have been established addressing the coordination with ASLBP to obtain stenographers for IITs, including provisions for off-duty hours and holidays.

5.4.5.4 Equipment

The IIS has the following equipment available for use by IIT members, along with controls for assigning and returning equipment: 35mm camera with flash and lens attachments, carrying case, and film supply; mini-cassette recorders with tapes and batteries; mini-transcribers; key cards; kastle cards; a portable computer; and a "go-bag" containing this equipment, procedures, and supplies for the IIT when in the field.

5.4.5.5 Onsite Support Arrangements

Procedures have been established to ensure that proper arrangements are made for the IIT while on site, including obtaining access to the site and making arrangements for providing the IIT with the physical facilities necessary to accomplish the onsite investigation efficiently and with a minimum of disruption to all involved. Procedures also address the proper recordkeeping of time and attendance of IIT members while onsite.

5.4.5.6 Records and Documentation Control

Procedures have been established to ensure that all information will be handled in a systematic manner to minimize the probability of lost information, to ensure that safeguards and proprietary information is properly controlled, and to ensure that information collected during the investigation is readily

retrievable. Specific guidance is provided for the handling of sensitive information such as transcripts and allegations of wrongdoing.

5.4.6 Development of the Incident Investigation Manual

The Incident Investigation Manual was prepared to provide guidelines for the conduct of investigative activities by the IITs. The guidelines were developed and coordinated by the IIS and reflect the experience gained from previous IIT investigations and other pertinent investigations. The Incident Investigation Manual addresses the specific points and concerns identified in the Commission Paper (SECY-85-208) and includes the following:

5.4.6.1 Activating an IIT

This provides guidance to NRC management for activating an IIT response and addresses the following areas:

- Selection and Scope of Events for IIT Response
- IIT Membership
- IIT Activation Process
- Participation by Industry Organizations
- Augmented Inspection Team (AIT) Response
- Upgrading or Downgrading an IIT

5.4.6.2 Conduct of Investigation

This provides guidance to the IIT leader for conducting an investigation including the following topics:

- Scope of the Investigation
- Team Leader Responsibilities
- Role of the Region
- Initial Actions by the Team Leader
- Entrance Meeting with the Licensee
- Plant Tour of Equipment and Systems
- Interviewing Personnel
- Sequence of Events
- Development of the Quarantine Equipment List (QEL)
- Responding to Press Inquiries
- IIT Coordination Meetings
- Identifying Additional Expertise and Outside Assistance
- Industry Participation in the Investigation
- Parallel Investigations
- Status Reports
- IIT Recordkeeping Activities
- Collection of Information
- Referral of Investigation Information to NRC Offices
- Confidentiality
- Subpoena Power and Power to Administer Oath and Affirmation
- IIT Investigation Sequence
- Return Site Visit
- Report Preparation and Presentation

5.4.6.3 Interviews

This provides guidance to IITs to ensure interviews are conducted in a uniform, systematic, and complete manner and addresses the following areas:

- Guidance for Conducting Interviews
- Handling of Transcripts and Access by Other Parties
- Attendance of Third Parties at Interviews

5.4.6.4 Treatment of Quarantined Equipment

This provides guidance for handling quarantined equipment and developing related troubleshooting action plans including the following areas:

- Licensee's Responsibilities
- Quarantined Equipment List (QEL) Guidelines
- Guidance for Developing Troubleshooting Action Plans

5.4.6.5 Report Preparation

This provides guidance for the preparation, release, and distribution of the IIT report including the following areas:

- Writing and Publishing Guidelines
- Report Writing Guidelines
- Graphic Guidelines
- Publication Forms
- Distribution of the Advance Copy
- Distribution of the Published NUREG
- Schedule

5.4.6.6 Background Information

This section contains background information for the IIP including:

- NRC Manual Chapter 0513, Incident Investigation Program
- IE Procedure for Augmented Inspection Team Response to Operational Events
- SECY-85-208 entitled "NRC Incident Investigation Program"

The manual was issued for review and comment on June 16, 1986, discussed during the training program, transmitted to the Owners' Groups, NSAC, and INPO for review on August 4, 1986, and issued for trial use on September 12, 1986.

5.4.7 Regional Workshops

During the last months of 1986, the IIS made preparations to hold a one-day workshop on the NRC's Incident Investigation Program in each of the regional offices. The purpose of the workshop is to acquaint utilities with the IIP and provide an understanding of the process by which NRC will investigate potential significant incidents. It also provides an opportunity for the utilities to be better prepared should an incident at one of their facilities trigger the establishment of an IIT or an AIT. The workshop provides a forum for questions to be raised and answered, and for industry comments to be incorporated into the next revision of the Incident Investigation Manual.

5.4.8 Industry Participation

The objective of this activity is to have representatives from outside organizations, such as INPO, NSAC, and utility Owners' Groups, participate fully as team members in future IITs. Their participation would add additional perspective to the investigation and expert knowledge of industry practices in numerous areas. Guidance concerning industry representatives on IITs was developed which addresses the qualifications, selection, responsibilities, and role of industry participants on IIT investigations. This guidance was forwarded and subsequently discussed in meetings with INPO, NSAC, and all utility Owners' Groups to elicit their comments concerning industry involvement in the IIT process. A number of issues raised during these discussions, including questions about an industry representative's liability, about NRC supervision of non-government members of an IIT, about the need for a statement of confidentiality from non-NRC IIT members, and about the application of the Federal Advisory Committee Act are being addressed. Resolution of these issues in coordination with the Office of General Counsel is expected in 1987.

6. STATUS OF AEOD RECOMMENDATIONS

This section summarizes the year-end status of all AEOD recommendations which are either new or still outstanding from the last report. The status of a total of 100 recommendations are provided in this section. These have been categorized (as of December 31, 1986) as follows:

Status of AEOD Recommendations

° Added Since Last Report	28
° Resolved Since Last Report	11
° Total Not Resolved	79
- Currently Under Review	17
- Proceeding Satisfactorily	59
- Not Proceeding Satisfactorily	3

For tracking purposes, "Resolved Since Last Report" means that formal NRC action has been taken requesting licensee action. "Currently Under Review" means that the recommendation is being assessed by another NRC office and no position has yet been taken by the responsible program office. "Proceeding Satisfactorily" means that the responsible program office has agreed with the recommendation and has initiated appropriate action. "Not Proceeding Satisfactorily" means that the recommendation was not accepted by the program office and additional discussion between the program office and AEOD is being or will be held. At this time, there are no issues involving AEOD recommendations which would warrant EDO attention. AEOD's recommendation tracking system ensures that all formal AEOD recommendations are tracked until resolution is achieved.

In addition to formal recommendations which are tracked and included in this section, additional actions are routinely implemented by NRC program offices on AEOD suggestions contained in engineering evaluations and special reports. These AEOD suggestions are not formally tracked or closed out, and no formal response is required. Section 5.1 highlights a number of examples of where action was initiated based upon AEOD suggestions.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C101

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "St. Lucie Natural Circulation Cooldown"

RECOMMENDATION 1

Provide a supply of cooling water to reactor coolant pump seals that will not be disabled by a single failure.
(Recommendation 4e)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/EIB

J. Jackson

High

6/87

Proceeding satisfactorily. This recommendation is included in Generic Issue 65, "Probability of Core Melt due to Component Cooling Water System Failures." Generic Issue 65 comprises task 2 of Generic Issue 23, "Reactor Coolant Pump Seal Failures" and will address the reliability of RCP seal cooling systems. During 1986 the technical studies and research needed for GI-23 were completed. The next step is for the staff to prepare a proposed resolution package for CRGR review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C105

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980"

RECOMMENDATION 1

Installation of dual atmospheric dump valve capability for each steam generator on two-loop PWRs.
(Study recommendation 8(b)3)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Marchese

High

4/87

Proceeding satisfactorily. This recommendation originally was to have been addressed by a revision to SRP 15.6.3 as part of Generic Issue 67.5.1, "Reassessment of Radiological Consequences Following a Postulated Steam Generator Tube Rupture." This recommendation may be addressed in USI A-45, "Shutdown Decay Heat Removal Requirements" depending on the final resolution of this issue. In the past year a draft regulatory analysis and backfit report that includes this recommendation as part of A-45 was prepared. This report is being reviewed by NRR prior to CRGR review.

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AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C105 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980"

RECOMMENDATION 2

Review of steam generator tube rupture (SGTR) analyses for plants licensed prior to the SRP. (Study recommendation 8(b)2)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/EIB

K. Shaukat

Medium

Not scheduled

Not proceeding satisfactorily. As a result of contacts made with cognizant NRR staff after January 1, 1987 it was learned that this issue was inadvertently left unassigned by the NRP staff during 1986. It was to have been included with USI A-47, "Safety Implications of Control Systems." Shortly after contact was made the recommendation was assigned a contact and it may now be addressed as part of GI-67, "Reevaluation of SGTR Design Basis Event." However, ongoing SGTR programs including GI-67 are being integrated into a single generic issue (GI-135). This recommendation may be included in the integrated program.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C105 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980"

RECOMMENDATION 3

Revise SRP 9.2.2 to clarify isolation of nonsafety-related portions of service water system.
(Study recommendation 8(a)6)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/SPEB

L. Riani

N/A

N/A

Resolved. Standard Review
Sections 9.2.1 and 9.2.2 were
revised to address this
recommendation. The revised
SRPs were issued in June 1986.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C105 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980"

RECOMMENDATION 5

IST of check valves in the instrument air system used to isolate safety-related portions of the system.
(Study recommendation 8(a)2)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/SPEB

W. Milstead

Low/drop

N/A

Not proceeding satisfactorily. This issue is to be addressed as part of Generic Issue 43, "Contamination of Instrument Air Lines" which was prioritized by NRR as low/drop. AEOD did not agree with this prioritization. NRR agreed to reevaluate the prioritization when a comprehensive AEOD case study on air system problems was issued. In December 1986 AEOD issued for peer review a preliminary case study on this subject. The final report is expected to be issued in early 1987 at which time we would expect NRR to reassess the priority of this recommendation.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum to Harold Denton from C. J. Heltemes, dated May 2, 1983

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: Response to NRR Comments on AEOD Report, "Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980"

RECOMMENDATION 1

Accessibility of ADVs for local manual operations for RCS cooldown following a steam generator tube rupture.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	A. Marchese	High	7/87	Proceeding satisfactorily. This recommendation is currently included in USI A-45. A site walkdown survey of nine plants substantiated the AEOD concern of ADV accessibility. At some plants, the ADVs were readily accessible, while ADVs at other plants were difficult to open manually and may cause personnel radiation exposure. In 1986 a draft regulatory and backfit report that addresses this recommendation has been prepared. This report is currently undergoing internal NRR staff review prior to being forwarded to CRGR.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum dated January 20, 1982 from C. Michelson to Harold R. Denton

Responsible AEOD Engineer: M. Chiramal/F. Ashe

TITLE OR SUBJECT: "Safety Concerns Associated with Reactor Vessel Level Instrumentation in BWRs"

RECOMMENDATION 1

Safety-related low-low reactor vessel level start of HPCI and RCIC systems should not be prevented or delayed by nonsafety-related high level trip.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	J. Joyce	High	Not scheduled	Proceeding satisfactorily. Assigned as Generic Issue 101.

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RECOMMENDATION 2

Protective functions of narrow range level instrumentation must be assured in spite of adverse control system protection system interaction.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/EIB	A. Szukiewicz	High	Not scheduled	Proceeding satisfactorily. On-going USI-A47 and Generic Issue 101.

REFERENCE: Memo dated March 19, 1982 from H. R. Denton to C. Michelson.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C202

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Flow Blockage by Bivalve Mollusks at Arkansas Nuclear One and Brunswick"

RECOMMENDATION 1

Capability to measure cooling water flow should be provided for all safety-related equipment.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW*</u>	<u>STATUS</u>
NRR/DSRO/EIB	C. Hickey	Medium	1/88	Proceeding satisfactorily. This recommendation has been included in Generic Issue 51, "Reliability of Open Cycle Service Water Systems." During 1986 a contractor study which includes this recommendation has been approved (FIN-B-2977), funded and initiated. The final contractor report is scheduled for completion by 10/87.

RECOMMENDATION 2

Develop and implement biofouling control strategies.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW*</u>	<u>STATUS</u>
NRR/DSRO/EIB	C. Hickey	Medium	1/88*	Proceeding satisfactorily. (See status of Recommendation 1 above.)

*May not be required. If necessary, will be scheduled following completion of Generic Issue 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems." Generic Issue 51 is incorporating Task V of Project FIN B-2977 by RES.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C202 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Flow Blockage by Bivalve Mollusks at Arkansas Nuclear One and Brunswick"

RECOMMENDATION 3

Periodic inspection of service water system piping.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW*</u>	<u>STATUS</u>
NRR/DSRO/EIB	C. Hickey	Medium	1/88	(See status of Recommendation 1 above.)

RECOMMENDATION 4

Periodic verification of overall heat transfer coefficient on multiple pass heat exchangers.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW*</u>	<u>STATUS</u>
NRR/DSRO/EIB	C. Hickey	Medium	1/88	(See status of Recommendation 1 above.)

*May not be required. If necessary, will be scheduled following completion of Generic Issue 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems." Generic Issue 51 is incorporating Task V of Project FIN B-2977 by RES.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C202 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Flow Blockage by Bivalve Mollusks at Arkansas Nuclear One and Brunswick"

RECOMMENDATION 5

Periodic verification of cooling water flow to all safety-related equipment should be specified in technical specifications.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW*</u>	<u>STATUS</u>
NRR/DSRO/EIB	C. Hickey	Medium	1/88	(See status of Recommendation 1 above.)

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*May not be required. If necessary, will be scheduled following completion of Generic Issue 51, "Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems." Generic Issue 51 is incorporating Task V of Project FIN B-2977 by RES.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C203 and Memorandum dated May 28, 1982.

Responsible AEOD Engineer: E. J. Brown

TITLE OR SUBJECT: "Survey of Valve Operator - Related Events Occurring During 1978, 1979, and 1980"
(See also Recommendation 1 on page S-503-1)

RECOMMENDATION 1

Existing guidance to bypass thermal overload protective devices associated with safety-related valve motor operators should be reassessed.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW*</u>	<u>STATUS</u>
122	NRR/DSRO/EIB	O. Rothberg	Medium	Not scheduled	Proceeding satisfactorily. This issue is being addressed in Generic Issue II.E.6, "In- Situ Testing of Valves," which is being evaluated under contract with BNL. This issue will also be further evaluated by NRR as part of their review of the final AEOD Case Study on MOV performance, which was issued in December 1986.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C203 and Memorandum dated May 28, 1982 (continued).

Responsible AEOD Engineer: E. J. Brown

TITLE OR SUBJECT: "Survey of Valve Operator - Related Events Occurring During 1978, 1979, and 1980"
(See also Recommendation 1 on page S-503-1)

RECOMMENDATION 2

Improved methods and procedures for the setting of torque switches should be developed and evaluated relative to valve operability and functional qualification.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/EIB

O. Rothberg

Medium

Not scheduled

Proceeding satisfactorily.
Based on a letter from the
Director, NRR dated 2/23/83,
an existing RES program was
expanded to address this sub-
ject. This has been included
in RES contract B3050, "Valve
Performance Testing."

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C203 and Memorandum dated May 28, 1982 (continued).

Responsible AEOD Engineer: E. J. Brown

TITLE OR SUBJECT: "Survey of Valve Operator - Related Events Occurring During 1978, 1979, and 1980"
(See also Recommendation 1 on page S-503-1)

RECOMMENDATION 3

Signature tracing techniques should be developed and tried on selected motor-operated valves as part of the periodic inservice testing program.

124

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CPGR
REVIEW

STATUS

NRR/DSRO/EIB
IE/DEPER/EGCB

O. Rothberg
R. Kiessel

Medium

Not scheduled

Proceeding satisfactorily. This was included as part of a proposed draft plan (not yet approved) for Generic Issue II.E.6, "In-Situ Testing of Valves." It will be covered by research programs. A user request memorandum, dated 5/14/84, from the Director, NRR to the Director, RES addressed this item. Valve testing was completed and a final report was issued in January 1986, as NUREG/CR-4380. Signature tracing is being used by many licensees in implementing IE Bulletin 85-03.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Case Study AEOD/C204 "San Onofre Unit 1 Loss of Salt Water Cooling Event on March 10, 1980, dated July 1982

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Single Failure Vulnerability of San Onofre 1's Salt Water Cooling System"

RECOMMENDATION 1

Ongoing efforts of the SEP focus on single failure vulnerability and consequences for the salt water cooling system and other equivalent service and cooling water systems.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DPLA/PDI

E. McKenna

High

N/A

Proceeding satisfactorily. NRR has reviewed SEP plants for such vulnerabilities. Modifications have been made at SEP plants as appropriate. San Onofre has made several modifications and is performing a reliability analysis of the modified salt water cooling system to confirm the adequacy of the system modifications. The analysis was scheduled for completion in August 1986, however, it is not yet completed and no new completion date has been received.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Memorandum, C. Michelson to Chairman Ahearne, "New Unresolved Safety Issues" dated August 4, 1980,
Memo, C. Michelson to H. Denton, "Resolution of Issue Concerning Steamline Break with Small LOCA,"
dated June 23, 1982, Case Study AEOD/C205, "ATOG as Applied to the April 1981 Overfill Event at ANO-1"

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Safety Implications of Steam Generator Transients and Accidents"

RECOMMENDATION 1

Combined primary/secondary side blowdown should be a USI for B&W plants.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DPLB/RSB
RES

R. Jones
B. Beckner

High

N/A

Proceeding satisfactorily.
B&W licensees/EPRI/NRC are
jointly funding a test
facility to obtain integral
systems test data to resolve
the uncertainties associated
with B&W plant response to
SBLOCA and other transients
and accidents.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Memorandum, C. Michelson to Chairman Ahearne, "New Unresolved Safety Issues" dated August 4, 1980,
Memo, C. Michelson to H. Denton, "Resolution of Issue Concerning Steamline Break with Small LOCA,"
dated June 23, 1982, Case Study AEOD/C205, "ATOG as Applied to the April 1981 Overfill Event at ANO-1"

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Safety Implications of Steam Generator Transients and Accidents" (continued)

RECOMMENDATION 2

TAP-A47 should focus on equipment modifications or additions to preclude SG overfill as a credible event.

RESPONSIBLE OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR REVIEW

STATUS

NRR/DSRO/EIB

A. Szukiewicz High

2/87

Proceeding satisfactorily.
Included in on-going Generic
Issue A-47 and the B&W Owners
Group reassessment. Most B&W
plants have committed to install
safety-grade systems to prevent
SG overfill protection. A CRGR
package, which includes a proposed
generic letter for resolution of
the issue, is currently being
reviewed by several NRC offices.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C301

Responsible AEOD Engineer: M. Chirama1

TITLE OR SUBJECT: "Failure of Class 1 E Safety-Related Switchgear Circuit Breakers to Close to Demand"

RECOMMENDATION 1

Provide for monitoring the status of the closing circuit of Class 1 E Circuit Breakers and for appropriately selected breakers such as diesel generator output breakers, make the status indication available to the control room operator.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/SPEB	K. Knie1	N/A	N/A	Proceeding satisfactorily. NRR is re-prioritizing Generic Issue 55 based on Reference 3.

RECOMMENDATION 2

In the short-term, licensees of operating reactors should establish regular local surveillance of Class 1 E switchgear circuit breakers to monitor the readiness status of the spring-charging motor of each unit.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/SPEB	K. Knie1	N/A	N/A	Proceeding satisfactorily. (See status of Recommendation 1 above.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C301 (continued)

Responsible AEOD Engineer: M. Chiramal

TITLE OR SUBJECT: "Failure of Class 1 E Safety-Related Switchgear Circuit Breakers to Close to Demand"

RECOMMENDATION 3

In addition to the above, measures that tend to preclude dirty or corroded contacts, poor electrical connections, blown control circuit fuses, and improper return of breakers to operable status should be incorporated into the maintenance procedures and used in actual maintenance practice.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/SPEB

K. Kniel

N/A

Proceeding satisfactorily.
(See status of Recommendation 1).

RECOMMENDATION 4

Shift operating personnel should receive periodic training in the logic and operation of circuit breakers equipped with anti-pumping controls.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/SPEB
IE

K. Kniel
V. Thomas

N/A

Proceeding satisfactorily.
(See status of Recommendation 1).
IE issued Information Notice 83-50
advising licensees of the problems.
This concern is being reconsidered
in NRR re-prioritization of the
issue.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Case Study AEOD/C301 (continued)

Responsible AEOD Engineer: M. Chirama1

TITLE OR SUBJECT: "Failure of Class 1 E Safety-Related Switchgear Circuit Breakers to Close to Demand"

REFERENCES:

- 1) Memo to C. H. Heltemes, Jr., from H. R. Denton, June 17, 1983 "AEOD April 1983 Report on Failure of Class 1 E Safety-Related Switchgear Circuit Breakers to Close on Demand"
- 2) Memo to D. G. Eisenhut from R. L. Spessard, June 1, 1984, "Unmonitored Failures of Class 1 E Safety-Related Switchgear Circuit Breakers to Close on Demand"
- 3) Memo to R. M. Bernero from H. R. Denton, March 27, 1985 "Scheduled for Resolving and Completing Generic Issue No. 55 - Failure of Class 1 E Safety-Related Switchgear Circuit Breakers to Close on Demand"
- 4) Memo to H. R. Denton from C. J. Heltemes, Jr., April 12, 1985 "Generic Issue No. 55 - Failure of Class 1 E Safety-Related Switchgear Circuit Breakers to Close on Demand"
- 5) Memo to C. J. Heltemes, Jr. from H. R. Denton, May 9, 1985 "AEOD Concerns Regarding Generic Issue No. 55"

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION SOURCE

Case Study AEOD/C401 and Memorandum from C. J. Heltemes, Jr. to H. Denton, dated March 16, 1984

Responsible AEOD Engineer: S. Salah

TITLE OR SUBJECT: "Low Temperature Overpressure Events at Turkey Point Unit 4"

RECOMMENDATION 1

Correct the LTOP technical specifications for the 5 areas identified in the report.

RESPONSIBLE OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR REVIEW

STATUS

NRR/DSRO/RSIB

Ed Throm

High

N/A

Proceeding satisfactorily. NRR has identified this activity as Generic Issue 94. NRR/DSRO has revised the prioritization to "High" based on our memorandum dated 6/3/85. Oak Ridge National Laboratory performed an LER search and provided the results to NRR (Letter from Mays to E. Throm dated Sept. 2, 1986). NRR has also signed a contract with PNL to perform some of the work related to this generic issue.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C402

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Operating Experience Related to Moisture Intrusion in Electrical Equipment at Commercial Power Reactors"

RECOMMENDATION 1

IE should revise the inspection program to ensure licensee adherence to NRC requirements.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

IE/DQASIP/ORPB

M. Johnson

None

Proceeding satisfactorily.
This recommendation is to be
addressed in five separate
maintenance procedure-related
IE inspection modules.
Regional comments have been
incorporated in these modules,
and are in the concurrence
chain for issuance.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C403

Responsible AEOD Engineer: P. Baranowsky

TITLE OR SUBJECT: "Edwin I. Hatch Unit 2 Plant Systems Interaction Event on August 25, 1982"

RECOMMENDATION 1

Evaluate the common mode failure potential of safety systems due to the harsh environment of breaks outside containment being back channelled through floor drain systems.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/EIB

D. Thatcher

High

3/87

Proceeding satisfactorily. This recommendation was originally to be evaluated as part of Generic Issue 77, "Backflow Protection in Common Equipment and Floor Drain Systems." In 1986 NRR consolidated GI-77 into A-17, "Systems Interactions." The staff prepared a proposed resolution package for A-17 which predominantly involves common mode flooding of nuclear plant equipment. The package was transmitted to all NRC offices and ACRS for comment and is currently being revised to address the comments received in preparation for CRGR review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C403 (continued)

Responsible AEOD Engineer: P. Baranowsky

TITLE OR SUBJECT: "Edwin I. Hatch Unit 2 Plant Systems Interaction Event on August 25, 1982"

RECOMMENDATION 2

Supplemental arrangements should be provided to assure timely isolation of the affected floor drain system if the results of the above evaluation result in unacceptable common-mode safety system failures.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/EIB

D. Thatcher

High

3/87

(See status of Recommendation 1).

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum from C. J. Heltemes, Jr. to H. Denton, dated July 23, 1984

Responsible AEOD Engineer: S. Israel

TITLE OR SUBJECT: "Steam Binding of Auxiliary Feedwater Pumps"

RECOMMENDATION 1

PWR licensees should establish a method to regularly monitor the AFW system to minimize the potential for steam binding.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB
IE/DEPER/EGCB

A. Spano
V. Hodge

High

TBD

Proceeding satisfactorily.
Identified as Generic Issue 93.
The Task Action Plan for GI 93
was issued on 1/22/85. The
present schedule for resolution
is mid-1987. IE Bulletin 85-01
has been issued. All PWR plants
are monitoring pipe temperature
every eight hours in the interim.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C405

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Breaching of the Encapsulation of Sealed Well Logging Sources"

RECOMMENDATION 1

Part 39 should require that well logging licensees have specific emergency procedures for handling specific source rupture incidents.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

RES/RAMBR

M. L. Ernst

N/A

Resolved. This recommendation was included in the final revised version 10 CFR 39.

RECOMMENDATION 2

Part 39 should require special physical characteristics for well logging sources.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

RES/RAMBR

M. L. Ernst

N/A

Resolved. This recommendation was included in the final revised version of 10 CFR 39.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C405 (continued)

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Breaching of the Encapsulation of Sealed Well Logging Sources"

RECOMMENDATION 3

Part 39 should preclude licensees from removing or attempting to remove sources from source holders without specific authorization in the license.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

RES/RAMBR

M. L. Ernst

N/A

Resolved. This recommendation was included in the final revised version of 10 CFR 39.

RECOMMENDATION 5

The interface between the well owner, well logging company, and companies specializing in recovery operations should be defined in order to establish the regulatory responsibility and authority over recovery and well logging operations.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

RES/RAMBR

M. L. Ernst

N/A

Resolved. This recommendation was included in the final revised version of 10 CFR 39.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C501 (NUREG/CR-3551)

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants"

RECOMMENDATION 2

Require protection to prevent hydrogen explosions or fires in areas containing or impacting operation of safety related equipment.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/SPEB

W. Milstead

Low

N/A

Proceeding satisfactorily. Generic Issue No. 106, "Piping and the Use of Highly Combustible Gases in Vital Areas" was prioritized by PNL for the staff and given a low priority. Due to increased hydrogen use proposed for BWRs to reduce oxygen in the recirculation system for IGSCC control, the priority of this issue will be re-evaluated by NRR.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C502

Responsible AEOD Engineer: P. Lam

TITLE OR SUBJECT: "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors"

RECOMMENDATION 1

Disable the nonsafety-related air operator associated with the testable isolation check valve on the injection line in the emergency core cooling systems.

139

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

H. Woods

High

1988

Proceeding satisfactorily. Assigned as Generic Issue 105 "Interfacing Systems LOCA at BWRs." BNL has been contracted to conduct an evaluation and risk analysis of the issue with the draft report due by May 1987.

RECOMMENDATION 2

Perform leakage testing of the testable isolation check valve prior to plant startup after each refueling outage or following maintenance, repair or replacement of the valve.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

H. Woods

High

1988

Proceeding satisfactorily. Assigned as Generic Issue 105. BNL has been contracted to conduct an evaluation and risk analysis of the issue with the draft report due by May 1987.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C502 (continued)

Responsible AEOD Engineer: P. Lam

TITLE OR SUBJECT: "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors"

RECOMMENDATION 3

Reduce human errors in maintenance and surveillance testing activities.

140

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

H. Woods

High

1988

Proceeding satisfactorily. Assigned as Generic Issue 105. "Interfacing Systems LOCA at BWRs." BNL has been contracted to conduct an evaluation and risk analysis of the issue with the draft report due by May 1987.

RECOMMENDATION 4

Study reducing the frequency of surveillance testing of the isolation barriers of the emergency core cooling systems during power operation.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

H. Woods

High

1988

Proceeding satisfactorily. (See status of Recommendation 3 above.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C503

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Decay Heat Removal Problems at U.S. PWRs"

RECOMMENDATION 1

NRR assess the need for NRC requirements to improve planning, coordination, procedures, and personnel training during shutdown to ensure the availability of the DHR system.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
141	NRR/DSRO/RSIB	A. Spano	High	N/A	Proceeding satisfactorily. Generic Issue 99, "RCS/RHR Suction Line Valve Interlock on PWRs" has been expanded to include this recommendation. NRR has contracted with BNL to perform a PRA on the effects of implementing this recommendation at the Zion plant. Methods of extrapolating the work for Zion to other plants are also to be studied. The study is to be completed by the spring of 1987.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C503 (continued)

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Decay Heat Removal Problems at U.S. PWRs"

RECOMMENDATION 2

NRR require PWR licensees to have a reliable method of measuring and monitoring reactor vessel level during shutdown modes of operation and corresponding technical specification requirements for operability.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Spano

High

N/A

Proceeding satisfactorily. (See
status of Recommendation 1.)

RECOMMENDATION 3

NRR require licensees to improve the man-machine interfaces related to DHR operation.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Spano

High

N/A

Proceeding satisfactorily. (See
status of Recommendation 1.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C503 (continued)

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Decay Heat Removal Problems at U.S. PWRs"

RECOMMENDATION 4

NRR should consider DHR suction bypass lines as alternatives to redundant drop lines (if A-45 concludes that single drop line configurations are unacceptable).

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Spano

High

N/A

Proceeding satisfactorily. (See status of Recommendation 1.)

RECOMMENDATION 5

NRR consider removal of autoclosure interlocks to minimize loss-of-DHR events.

RESPONSIBLE
OFFICE/DIV/RSIB

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Spano

High

N/A

Proceeding satisfactorily. (See status of Recommendation 1.)

RECOMMENDATION 6

NRR should address the issue of DHR system redundancy to ensure that the DHR system is available during Mode 4, and the early stages of Mode 5.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Spano

High

N/A

Proceeding satisfactorily. (See status of Recommendation 1.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C504

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Loss of Safety System Function Events"

RECOMMENDATION 1

IE should issue an information notice to feed back the results of the case study to industry.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

IE/DEPER/EGCB

R. Baer

N/A

N/A

Completed. IE determined and document by memorandum dated February 1987 that an Information Notice (IN) would not be an appropriate vehicle for feeding back to industry the results of the case study. This was because the wide variety of systems and problems covered in the case study would not permit recommending corrective action(s) and because IE has issued INs and Bulletins on specific problems that have occurred in systems covered in the case study. During 1986 and 1987, two IE Bulletins, 18 INs, and one IN

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C504 (continued)

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Loss of Safety System Function Events"

RECOMMENDATION 1

IE should issue an information notice to feed back the results of the case study to industry.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGP REVIEW</u>	<u>STATUS</u>
145	IE/DEPER/EGCB	R. Baer		.	(continued) supplement have been issued to address different facets of this subject. AEOD also distributed the case study report to INPO, NSAC, and the Owners' Groups, and to the licensees that had experienced the events. In addition, the Performance Indicator program has included safety system failures as one of the six indicators for quarterly review. Finally, the results of the study were published in a November 1986 study by the CSNI Principal Working Group #1 (Operating Experience and Human Factors) report entitled "Loss of Safety System Functions."

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C504 (continued)

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Loss of Safety System Function Events"

RECOMMENDATION 2

NRR review the Maintenance and Surveillance Program Plan, the Human Factors Program Plan, and the INPO training accreditation program to ensure the adequacy of training programs for all types of NPP personnel.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DHFT/HFIB

D. Jones

Low

N/A

Not proceeding satisfactorily. Understaffing and other higher priority HFIB projects have delayed work on this item.

RECOMMENDATION 3

AEOD determine whether or not to perform further evaluations of losses of ECCS injection systems events, containment spray isolation events, and the Salem loss of component cooling water event.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

AEOD

E. Trager

N/A

N/A

Resolved. The San Onofre 1 and Beaver Valley 1 Loss of ECCS injection systems events were evaluated in NUREG/CR-3591, "Precursors to Potential Severe Core Damage Accidents: 1980-1981,

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C504 (continued)

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Loss of Safety System Function Events"

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

AEOD

E. Trager

N/A

N/A

(continued)

A Status Report," where they were assigned estimated conditional core damage probabilities of 1E-4 and 2E-6, respectively. A similar analysis using the Accident Sequence Precursor (ASP) evaluation techniques was performed on the remaining four events and they were found to have a low to moderate potential for core damage. The four estimated conditional core damage probabilities ranged from 1E-6 to 3E-5, and were smaller than the earlier San Onofre 1 event. The monitoring of containment spray isolation events since issuance of AEOD/C504 has not indicated the need for a specific study on this topic. The loss of component cooling water event at Salem has been included in an ongoing AEOD study entitled "Failures of Service Water System at Westinghouse Plants from 1981 to 1985."

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C505

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Therapy Misadministrations Reported to the NRC Pursuant to 10 CFR 35.42"

RECOMMENDATION 1

The Office of Nuclear Material Safety and Safeguards should communicate the information contained in this report to the affected licensees.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NMSS

R. Cunningham

N/A

N/A

Resolved. Copies of the case study report were transmitted to affected licensees in December 1986 by a letter summarizing the report findings.

RECOMMENDATION 2

The Office of Nuclear Material Safety and Safeguards should consider the following actions in regards to establishing quality assurance requirements for radiotherapy facilities licensed by NRC:

- Contact appropriate professional organizations to encourage and support the initiation of a voluntary, industry-directed physical quality assurance program for radiotherapy facilities. We believe that the commitment of the profession organizations in this regard should be assessed by the NRC and a conclusion reached as to the effectiveness of the voluntary program within two years.
- If substantial progress toward completion of the voluntary program, including a final completion date, has not been demonstrated at the end of two years, we recommend that NMSS initiate the necessary studies to determine whether a rulemaking is justified to require that radiotherapy facilities licensed by NRC have quality assurance programs to insure the accuracy of patient doses. The program should include such things as: independent

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C505 (continued)

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Therapy Misadministrations Reported to the NRC Pursuant to 10 CFR 35.42"

verification of patient does calculations and independent verification of the activity of brachytherapy sources before the sources are implanted.

-- The voluntary quality assurance program should contain at least the elements outlined above.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NMSS/FCMC

R. Cunningham

N/A

N/A

Proceeding satisfactorily. NMSS has undertaken rulemaking in regard to requiring quality assurance programs for radiotherapy facilities. An ANPRM is scheduled to be submitted to the Commission in early 1987. It appears that this rulemaking will satisfy our concerns as stated in Recommendation 2 in regard to quality assurance for radiotherapy facilities. We will continue to monitor the progress and content of the rulemaking.

149

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C505 (continued)

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Therapy Misadministrations Reported to the NRC Pursuant to 10 CFR 35.42"

RECOMMENDATION 3

10 CFR Part 35.21 should be amended to include the calibration of beam modifiers such as wedge filters, shaping filters, trays etc.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NMSS/FCMC

R. Cunningham

N/A

N/A

Proceeding satisfactorily. An ANPRM is being submitted to the Commission in early 1987. This ANPRM includes Recommendation 3. We will continue to monitor the progress and content of the proposed rulemaking.

RECOMMENDATION 4

In addition, to the extent that the NRC implements recommendation 3, the action should be made an item of compatibility for Agreement States.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

SP

W. Kerr

N/A

N/A

Proceeding satisfactorily. (See status of Recommendation 3.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C601

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Rupture of an Iodine-125 Brachytherapy Source at the University of Cincinnati Medical Center"

RECOMMENDATION 1

IE should publish an information notice describing the event and actions taken by the licensee and manufacturer to prevent recurrence.

151

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

CRGR
PRIORITY

REVIEW

STATUS

IE/DI/SMPB

H. Karagiannis

N/A

N/A

Resolved. IE IN-86-84 was
issued on September 30, 1986.

RECOMMENDATION 2

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NMSS/FC/FCML

K. Smith

N/A

N/A

Proceeding satisfactorily.
NMSS is working with Region
III to get 3M to provide
improved guidance to licensees.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C601 (continued)

Responsible AEOD Engineer: S. Pettijohn

TITLE OR SUBJECT: "Rupture of an Iodine-125 Brachytherapy Source at the University of Cincinnati Medical Center"

RECOMMENDATION 3

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NMSS/FC/FCML

K. Smith

N/A

N/A

Proceeding satisfactorily.
NMSS is evaluating the
prerogatives of broad licensees
and the context of the
recommendation to determine to
what extent the recommendation
might be implemented.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C602

Responsible AEOD Engineer: C. Hsu

TITLE OR SUBJECT: "Operational Experience Involving Turbine Overspeed Trips"

RECOMMENDATION 1

Alleviate the effect of slow response of the governor valve during the turbine startup transient by implementing a steam bypass modification for turbines equipped with a Woodward Model EG governor.

153

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	S. Diab	High	N/A	Proceeding satisfactorily. Included in on-going Generic Issues 122 and 124.

RECOMMENDATION 2

Avoid entrapped oil in the speed setting cylinder by establishing administrative controls for bleeding off the entrapped oil, installing a controllable dump valve, or providing indication in the control room for a spinning turbine.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	S. Diab	High	N/A	Proceeding satisfactorily. Included in on-going Generic Issues 122 and 124.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C602 (continued)

Responsible AEOD Engineer: C. Hsu

TITLE OR SUBJECT: "Operational Experience Involving Turbine Overspeed Trips"

RECOMMENDATION 3

Ensure the adequacy of the existing vendor-supplied calibration procedures for governor speed setting.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	S. Diab	High	N/A	Proceeding satisfactorily. Included in on-going Generic Issues 122 and 124.

RECOMMENDATION 4

Prevent water induction into the turbine by providing adequate provisions for condensate removal from the steam supply line to the AFW turbine.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	S. Diab	High	N/A	Proceeding satisfactorily. Included in on-going Generic Issues 122 and 124.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C602 (continued)

Responsible AEOD Engineer: C. Hsu

TITLE OR SUBJECT: "Operational Experience Involving Turbine Overspeed Trips"

RECOMMENDATION 5

Minimize trip and reset problems by assessing the adequacy of the existing procedural instruction, upgrading the training program, and providing local indication as well as control room indication for trip and reset conditions.

155

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/RSIB	S. Diab	High	N/A	Proceeding satisfactorily. Included in on-going Generic Issues 122 and 124.

RECOMMENDATION 6

Issue an information notice to alert licensees of operating reactors of the findings that led to the above recommendations.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
IE/DEPER/EGCB	J. Henderson	High	N/A	Resolved. Information Notice IE IN-86-14, Supplement 1 was issued on December 17, 1986.

REFERENCE: Memo dated October 22, 1986 from H. R. Denton to C. J. Heltemes, Jr.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C603

Responsible AEOD Engineer: E. Brown

TITLE OR SUBJECT: "A Review of Motor-Operated Valve Performance"

RECOMMENDATION 1

Expediently implement the recommendations in AEOD/C203 (May 1982) and AEOD/S503 (September 1985).

156

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NUMARC	Not assigned	N/A	N/A	Currently under review. AEOD Case Study C603 was transmitted to NUMARC by Reference 1 requesting that they take actions necessary to implement the recommendations in the case study. In Reference 2 NUMARC agreed to initiate a program to address the concerns and recommendations.

REFERENCE: 1. Letter from V. Stello, NRC to Warren Owen, NUMARC dated December 10, 1986
2. Letter dated January 7, 1987 to V. Stello from W. H. Owen

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C603 (continued)

Responsible AEOD Engineer: E. Brown

TITLE OR SUBJECT: "A Review of Motor-Operated Valve Performance"

RECOMMENDATION 2

Require licensees to establish procedures and diagnostic capability to determine root causes of MOV failure to operate.

157

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NUMARC	Not assigned	N/A	N/A	Currently under review. (Same as recommendation 1 above.)

RECOMMENDATION 3

Require licensees to develop a strong training program to ensure appropriate information and instructions are disseminated to operating and maintenance personnel.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NUMARC	Not assigned	N/A	N/A	Currently under review. (Same as recommendation 1 above.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C603

Responsible AEOD Engineer: E. Brown

TITLE OR SUBJECT: "A Review of Motor-Operated Valve Performance"

RECOMMENDATION 4

The scope of IE Bulletin 85-03 should be extended to cover all safety-related MOV assemblies required to be tested in accordance with 10 CFR 50.55a(g).

158	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
	NUMARC	Not assigned	N/A	N/A	Currently under review. (Same as recommendation 1 above.)

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C604

Responsible AEOD Engineer: M. Chirama1

TITLE OR SUBJECT: "Effects of Ambient Temperature on Electronic Components in Safety-Related I&C Systems"

RECOMMENDATION 1

Procedures for (a) loss of HVAC systems supplying instrumentation and control system equipment rooms and areas, and (b) loss of forced cooling to instrument cabinets, should be provided.

159

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR

Not assigned

N/A

N/A

Currently under review.

RECOMMENDATION 2

Training of control room and plant operators in using the procedures should be provided.

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR

Not assigned

N/A

N/A

Currently under review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C604 (continued)

Responsible AEOD Engineer: M. Chirama1

TITLE OR SUBJECT: "Effects of Ambient Temperature on Electronic Components in Safety-Related I&C Systems"

RECOMMENDATION 3

Supplemental cooling equipment should be readily available and identified for use in the event (of a loss control room cooling).

160

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR	Not assigned	N/A	N/A	Currently under review.

RECOMMENDATION 4

Periodically measure or continuously monitor the environmental conditions inside the instrument cabinets that contain heat sensitive solid-state components.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR	Not assigned	N/A	N/A	Currently under review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C604 (continued)

Responsible AEOD Engineer: M. Chirama

TITLE OR SUBJECT: "Effects of Ambient Temperature on Electronic Components in Safety-Related I&C Systems"

RECOMMENDATION 5

The room ambient temperature limit specified in the plant technical specifications for operability of the control room cooling and ventilation systems, should reflect the actual measured temperatures in the safety-related instrumentation and control system cabinets located in the control room area.

161

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR	Not assigned	N/A	N/A	Currently under review.

RECOMMENDATION 6

In the on-going plant-specific evaluations associated with the resolution of USI A-44, Station Blackout, the following considerations regarding the effects of high ambient temperature on solid-state electronic components should be included:

- (a) The design adequacy should be evaluated for instrumentation and control system equipment needed to function during and recovering from a station blackout, as well as other equipment whose malfunction would impact operability of such equipment.
- (b) Plant-specific equipment qualification data should be required unless the equipment qualification data can be verified by actual measurements of as-built and as-installed conditions.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C604 (continued)

Responsible AEOD Engineer: M. Chirama1

TITLE OR SUBJECT: "Effects of Ambient Temperature on Electronic Components in Safety-Related I&C Systems"

RECOMMENDATION 6 (continued)

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
162	NRR	Not assigned	N/A	N/A	Currently under review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C605

Responsible AEOD Engineer: F. Ashe

TITLE OR SUBJECT: "Operational Experience Involving Losses of Electrical Inverters"

RECOMMENDATION 1

The Office of Inspection & Enforcement (IE) should issue an information notice which addresses events involving inverter losses.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
163	IE/EGCB	V. Thomas	Medium	None	Proceeding satisfactorily. An IE IN is in the concurrence chain for issuance.

RECOMMENDATION 2

The circuitry which monitors the position of the circuit breakers for reactor coolant pump motors should be reassessed by NRR.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
	NRR	Not assigned	N/A	N/A	Currently under review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Case Study AEOD/C605 (continued)

Responsible AEOD Engineer: F. Ashe

TITLE OR SUBJECT: "Operational Experience Involving Losses of Electrical Inverters

RECOMMENDATION 3

Technical specifications which specifically address inverters and/or attendant buses for comparable plant designs should be reviewed to ensure that action statements addressing plant operating restrictions are consistent.

164

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR

Not assigned

N/A

N/A

Currently under review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Memo: C. Michelson to R. Mattson, "NRC Action Plan Developed as a Result of TMI-2 Accident - Draft 3, Task III.E.3 Decay Heat Removal," April 24, 1980.

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Reliability of DHR Systems"

RECOMMENDATION 1

Reliability of DHR systems should be reviewed and where necessary upgraded on an expedited basis.

165

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/RSIB

A. Marchese

High

3/87

Proceeding satisfactorily.
Included in TAP A-45. NRR is
presently drafting its
position on this issue based
upon several contractor
studies.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum dated July 15, 1980 from C. Michelson to H. R. Denton

Responsible AEOD Engineer: M. Chirama1

TITLE OR SUBJECT: "Operational Restrictions for Class 1E 120 VAC Vital Instrument Buses"

RECOMMENDATION 1

Impose Technical Specification requirements concerning operational restrictions for Class 1E 120 VAC Vital Buses on operating plants.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
166	NRR/DSRO/EIB	D. Thatcher	High	Not scheduled	Proceeding satisfactorily. Currently being pursued under Generic Issue 48.

REFERENCES: 1. Memo dated September 29, 1980 from H. Denton to C. Michelson
2. Memo dated October 25, 1983 from H. Denton to D. Eisenhut

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memo dated August 27, 1980 from C. Michelson to H. R. Denton

Responsible AEOD Engineer: M. Chiramal

TITLE OR SUBJECT: "Tie Breaker Between Redundant Class 1E Buses - Point Beach Nuclear Plant Units 1 & 2"

RECOMMENDATION 1

Interconnection between redundant safety-related electrical load groups should comply with requirements of Regulatory Guide 1.6.

167	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
	NRR/DSRO/EIB	D. Thatcher	Medium	Not scheduled	Proceeding satisfactorily. Generic Issue 49 has been established for this issue.

REFERENCE: Memo dated October 16, 1980 from H. R. Denton to C. Michelson

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum to H. R. Denton from C. Michelson, dated January 19, 1981

Responsible AEOD Engineer: E. Brown

TITLE OF SUBJECT: "Degradation of Internal Appurtenances in LWRs"

RECOMMENDATION 1

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
168	NRR/DSRO/SPEB	H. Vander Molen	Low	N/A	Proceeding satisfactorily. Agreement between NRR and AEOD to include evaluation of loose parts as possible missiles during SG blowdowns has been reached. Additionally, it was agreed that AEOD would conduct a data search to identify loose parts affecting ESF system reliability to assist NRR in evaluating ESF system concerns (Reference). Results of an updated data search will be compiled and forwarded to NRR in early 1987.

REFERENCE: Memo, K. V. Seyfrit to W. Minners, February 15, 1985, "Evaluation of Generic Issue No. 35, Degradation of Internal Appurtenance in LWRs"

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum from C. Michelson to V. Stello and H. Denton, "Immediate Action Memo: Common Cause Failure Potential at Rancho Seco - Dessicant Contamination of Air Lines," September 15, 1981

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Plant Air Systems"

RECOMMENDATION 1

Obtain licensees' experience and assessment of this problem and determine course of corrective action if required.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
169	NRR/DSRO/SPEB	W. Milstead	Low/Drop	N/A	Not proceeding satisfactorily. NRR originally prioritized this issue (Generic Issue 43) and recommended that it be dropped. AEOD did not agree with the prioritization, and sent a memo (Reference) requesting NRR to hold final disposition of this issue in abeyance until a comprehensive AEOD report was written on this subject. In December 1986, AEOD issued for peer review a preliminary case study on this subject. The final report is expected to be

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Memorandum from C. Michelson to V. Stello and H. Denton, "Immediate Action Memo: Common Cause Failure Potential at Rancho Seco - Dessicant Contamination of Air Lines," September 15, 1981 (continued)

Responsible AEOD Engineer: H. Ornstein

TITLE OR SUBJECT: "Plant Air Systems"

RECOMMENDATION 1 (continued)

Obtain licensees' experience and assessment of this problem and determine course of corrective action if required.

170

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/SPEB

W. Milstead

Low/Drop

N/A

(continued)
issued in early 1987 at which
time NRR will reassess the
priority of this recommendation.

REFERENCE: Memo from C. J. Heltemes, Jr., to H. R. Denton, "Contamination of Instrument Air Lines,"
December 14, 1983

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum from C. Michelson to R. Vollmer and R. Mattson dated February 24, 1982

Responsible AEOD Engineer: M. Chiramal

TITLE OR SUBJECT: "Spurious Trip of the Generator Lockout Relay Associated with a Diesel Generator Unit"

RECOMMENDATION I

Should explicitly verify that seismic qualification of all protective devices used in the control and protection circuitry of DG units has been performed with these devices in their energized, de-energized, tripped and non-tripped states.

171

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DSRO/EIB

T. Chang

High

Not scheduled

Proceeding satisfactorily.
Issue has been incorporated
into USI A-46.

REFERENCE: Memorandum from H. R. Denton to C. Michelson dated May 11, 1982

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Engineering Evaluation AEOD/E215

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Salt Water System Flow Blockage at Pilgrim NPS by Blue Mussels"

RECOMMENDATION 1 Internal inspection of RBCCW HX supply headers

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW*

STATUS

NRR/DSRO/EIB

C. Hickey

Medium

1/88

Proceeding satisfactorily.
(See status of recommendation
1 on page C-202-1.)

RECOMMENDATION 2 Periodic measurement of overall heat transfer coefficient on RBCCW HXs at Pilgrim

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW*

STATUS

NRR/DSRO/EIB

C. Hickey

Medium

1/88

Proceeding satisfactorily.
(See status of recommendation
1 on page C-202-1.)

*May not be required. If necessary, will be scheduled following completion of Generic Issue 51, "Proposed Requirement for Improving the Reliability of Open Cycle Service Water Systems."

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Engineering Evaluation AEOD/E215 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Salt Water System Flow Blockage at Pilgrim NPS by Blue Mussels"

RECOMMENDATION 3 Periodic measurement of Salt Water System flow to RBCCW HXs

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW*

STATUS

NRR/DSRO/EIB

C. Hickey

Medium

1/88

Proceeding satisfactorily.
(See status of recommendation
1 on page C-202-1.)

173

*May not be required. If necessary, will be scheduled following completion of Generic Issue 51, "Proposed Requirement for Improving the Reliability of Open Cycle Service Water Systems."

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Engineering Evaluation AEOD/E304

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Backflow Protection in Common Equipment and Floor Drain Systems"

RECOMMENDATION 1

Provide backflow protection for drain systems in older operating plants.

174	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
	NRR/DSRO/EIB	D. Thatcher	High	Early 1987	Proceeding satisfactorily. This recommendation was initially prioritized as high and was to be included in Generic Issue 77, "Backflow Protection in Common Equipment and Floor Drain Systems." In 1986 NRR consolidated GI-77 into A-17, "Systems Interaction." The staff prepared a proposed resolution package for A-17 which principally involves common mode flooding of nuclear plant vital equipment spaces.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Engineering Evaluation AEOD/E304 (continued)

Responsible AEOD Engineer: T. Cintula

TITLE OR SUBJECT: "Backflow Protection in Common Equipment and Floor Drain Systems"

RECOMMENDATION 1 (continued)

Provide backflow protection for drain systems in older operating plants.

175

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

(continued)

NRR/DSRO

D. Thatcher

High

Early 1987

The package was transmitted to all NRC offices and ACPS for comment and is currently being revised to address the comments received in preparation for CRGR review.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report - C. J. Heltemes, Jr. to H. R. Denton dated January 13, 1984 and follow-up memorandum dated August 8, 1984

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Human Error in Events Involving Wrong Unit or Wrong Train"

RECOMMENDATION 1

Consider the need for further clarification or guidance on what constitutes an acceptable independent verification program.

176

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DHFT/MTB	G. Cwalina	High	N/A	Proceeding satisfactorily. (See status of recommendation 2 below.) The reference stated that "The NRC should provide clarifying guidance regarding independent verification." NRR plans to request that RES prepare regulatory guidance on activities such as independent verification programs that help to minimize the potential for the occurrence of these

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Special Study Report - C. J. Heltemes, Jr. to H. R. Denton dated January 13, 1984 and follow-up memorandum dated August 8, 1984 (continued)

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Human Error in Events Involving Wrong Unit or Wrong Train"

RECOMMENDATION 1 (continued)

RESPONSIBLE OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR REVIEW

STATUS

NRR/DHFT

G. Cwalina

High

N/A

(continued)
types of events. Action on this recommendation will be considered complete when the NRR request has been forwarded to RES. AEOD will continue to monitor the rate of occurrence of these types of events.

REFERENCE: "An Investigation of the Contributions to Wrong Unit or Wrong Train Events," NUREG-1192, April 1986.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report - C. J. Heltemes, Jr. to H. R. Denton dated January 13, 1984 and follow-up memorandum dated August 8, 1984 (continued)

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Human Error in Events Involving Wrong Unit or Wrong Train"

RECOMMENDATION 2

NRR review wrong unit/wrong train events and develop appropriate guidance to minimize such events.

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<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DHFT/MTB	G. Cwalina	High	N/A	Proceeding satisfactorily. NRR and AEOD staff completed a study of the factors that appeared to have contributed to these types of events. In April 1986 the results of the study were published in NUREG-1192 and were forwarded to INPO and NUMARC for consideration by industry. [In a meeting on August 15, 1986 with DHFT, NUMARC described industry efforts (e.g., procedure and labeling improvements) to

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE:

Special Study Report - C. J. Heltemes, Jr. to H. R. Denton dated January 13, 1984 and follow-up memorandum dated August 8, 1984 (continued)

Responsible AEOD Engineer: E. Trager

TITLE OR SUBJECT: "Human Error in Events Involving Wrong Unit or Wrong Train"

RECOMMENDATION 2 (continued)

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RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

NRR/DHFT/MTB

G. Cwalina

High

N/A

(continued)
reduce the occurrence of these types of events.] The results of the study were also considered while developing site visit protocols for use in Phase II of the MSPP during which industry actions to resolve the noted problems will be monitored. NRR has not been satisfied with the progress being made by industry and plans to request that RES provide regulatory guidance in this area. Action on this recommendation will be considered complete when the NRR request has been issued to RES. AEOD will continue to track the rates of occurrence of these types of events.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum - C. J. Heltemes, Jr., to R. DeYoung, May 11, 1984

Responsible AEOD Engineer: S. Rubin

TITLE OR SUBJECT: "Pressure Locking of Flexible Disk Wedge Type Gate Valves"

RECOMMENDATION 1

Give immediate consideration to issuing a bulletin on the subject of pressure locking of flexible disk wedge-type gate valves.

180

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

IE/DEPER

R. Singh

High

N/A

Resolved. IE initially prepared a draft bulletin as part of a CRGR review package. Shortly thereafter, it was learned that INPO had begun work on an SOER on this subject. AEOD and IE agreed to allow INPO to address the concerns, and activity by IE on the bulletin was suspended with AEOD agreement. On 12/14/84, INPO issued SOER 84-7, "Pressure Locking and Thermal Binding of Gate Valves." This document

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Memorandum - C. J. Heltemes, Jr., to R. DeYoung, May 11, 1984

Responsible AEOD Engineer: S. Rubin

TITLE OR SUBJECT: "Pressure Locking of Flexible Disk Wedge-Type Gate Valves"

RECOMMENDATION 1 (continued)

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RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

IE/DEPER

R. Singh

High

N/A

STATUS

(continued)

was found by AEOD and IE to be an acceptable substitute for the bulletin originally proposed. IE and AEOD have monitored the LERs, and 50.72 reports for adequacy of corrective actions taken in response to the INPO SOER. A review of the 1986 LER and NPRDS data related to this issue indicates that the SOER has been effective in addressing this issue and therefore no further NRC actions are required.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S503 and memorandum, C. J. Heltemes, Jr. to H. Denton dated September 19, 1985. (See also recommendations related to AEOD/C203 on pages C203-1 and C203-2).

Responsible AEOD Engineer: E. Brown

TITLE OR SUBJECT: "Evaluation of Recent Valve Operator Motor Burnout Events"

RECOMMENDATION 1

In view of the more than 200 motor burnout events, the NRR plan to address motor burnout should be expedited.

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<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
NRR/DSRO/EIB	O. Rothberg	Medium	N/A	Proceeding satisfactorily. This issue will be addressed in Generic Issue II.E.6, "In-Situ Testing of Valves," which is being evaluated under contract with BNL. As of mid-April 1986, the contract with BNL had been funded, approved and initiated. This issue will also be further evaluated by NRR during their review of the final AEOD Case Study on MOV performance which was issued in December 1986.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 1

- 183
- (1) IE should consider issuing an Information Notice (IN) to licensees to communicate the conclusions from this AEOD study and to suggest that licensees review their operating experience (OE) activities for effectiveness and for the intent and scope of conformance with the existing NRC requirements. The IN should note that licensees and the industry have lead responsibility for ensuring that the OE review activities are effective and reemphasize that the NRC considers the OE activities important to the safety of operations. The findings from the three IIT reports can be cited in this regard. Specifically, the IN should suggest that licensees, consistent with existing NRC requirements:
 - (a) Emphasize root cause analysis and implementation of permanent corrective actions to prevent recurrence of safety-related in-house events,
 - (b) Be diligent in the screening and assessment of industry feedback for applicability to their plant and in implementing corrective actions when appropriate,
 - (c) Disseminate information to non-supervisory plant operators, trades personnel, technicians and other support staff both directly and through training and retraining to keep them aware of anomalous events and their actual or potential consequences, and
 - (d) Implement self-monitoring of the effectiveness of their OE review activities through management reviews.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602 (continued)

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 1 (continued)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

IE/EPER/EGCB

V. Hodges

On hold

N/A

In July 1986, an IN was drafted that addressed S602 findings and conclusions, and an IE contract effort was being initiated to determine options for improving generic communications with licensees. The IE contract was let with completion due late in FY1987. The IN was not issued.

The IE contract includes interviews with NRC Headquarters and Regional Staff, licensees, INPO and others.

Further actions are pending completion of the contract.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602 (continued)

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 2

AEOD should have further discussions with the industry on additional initiatives, such as on means to increase the effectiveness of OE review activities, and on assessing the effectiveness of the OE review activities.

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<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
AEOD/PTB	J. Crooks	Continuing Activities	N/A	Met with INPO in February 1987 to discuss their initiatives in the area. One initiative was focused on the implementation of priority corrective actions from SOERs. Other improvements were made to program guidance and the evaluation techniques for assessing utility OE programs. PTB also included discussions on licensee's OE programs during plant visits related to the "new plant" study. In addition, discussions were

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602 (continued)

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 2 (continued)

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
AEOD/PTB	J. Crooks	Continuing	N/A	held with licensees who had read the report and wanted more information. Future plans are to visit additional licensee facilities and to meet further with INPO.

RECOMMENDATION 3

The NRC should continue to monitor operating experience and industry performance in OE activities for some time (2 years or so). If, after this period of time, sufficient improvement in effectiveness has not been observed, then further regulatory action would be appropriate. In this regard, AEOD believes the following actions should be initiated at this time:

- (a) NRR, in conjunction with AEOD and IE, should reappraise how the operating experience feedback activities might be improved through further guidance. The guidance could address the shortcomings noted in this study. The elements of an acceptable program could be defined in sufficient detail to permit determinations of program adequacy. Further, the guidance could address that both in-house and industry-wide experience are to be thoroughly assessed; that the applicable lessons are to be fed back to the various non-supervisory

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602 (continued)

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 3 (continued)

operating staff through direct feedback information systems and training programs; and that corrective actions are to be taken through procedure changes, training, and plant modifications. The guidance developed should, however, not be overly prescriptive, and should be consistent with present requirements.

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- (b) Based upon the above guidance, IE should develop a draft inspection module for reviewing the effectiveness of licensee practices for implementing operating experience feedback activities, including verification by NRC on a plant by plant basis (at least annually).
 - (c) IE should consider revising current OE feedback reviews to routinely include a sampling of industry feedback documents.

RESPONSIBLE OFFICE/DIV/BR

AEOD/PTB
NRR/ORAS
IE/DI/ORPB

CONTACT

J. Crooks
F. Brenneman
P. McKee

PRIORITY

Various

CRGR REVIEW

N/A

STATUS

- ° All groups are monitoring the OE programs as part of their normal functions.
- ° NRR, AEOD, and IE are awaiting the results of the IE contract for options for improvements to generic

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602 (continued)

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 3 (continued)

RESPONSIBLE
OFFICE/DIV/BR

CONTACT

PRIORITY

CRGR
REVIEW

STATUS

AEOD/PTB
NRR/ORAS
IE/DI/ORPB

J. Crooks
F. Brenneman
P. McKee

Various

N/A

communications, prior to
developing additional
guidance.

- ° IE has modified several procedures to better address the effectiveness of OE feedback.

RECOMMENDATION 4

Regarding OE feedback, as soon as practical, NRR, IE, and AEOD should consider consolidating existing systems (e.g., Generic Letters that deal with operational events, Information Notices that discuss operational events, Power Reactor Events) for transmitting operational experience into a single "NRC Notice" system. This "notice" would constitute the sole system for the feedback of operating experience to licensees for appropriate action and closeout. This system could specify the initiating office and the purpose (i.e., OE feedback), provide an indication of the significance and of the priority of attention expected, and could be sent to a single standard distribution list. In addition, the system could make sufficient information available for licensees to fully understand the concerns for assessment purposes and to conduct training when appropriate. In light of several licensee suggestions, consideration should be given to transmitting these "notices" via an electronic mail system.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S602 (continued)

Responsible AEOD Engineer: J. L. Crooks

TITLE OR SUBJECT: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

RECOMMENDATION 4 (continued)

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
AEOD/PTB NRR/ORAS IE/EGCB	J. Crooks F. Brenneman V. Hodges	On hold	N/A	Further actions are pending the results of the IE contract for options on improvements to generic communications.

RECOMMENDATION 5

In conjunction with the implementation of recommendation (4) above, AEOD should: (1) terminate the publication of Power Reactor Events, and (2) terminate the publication of the monthly Licensee Event Report (LER) Compilation, unless there are conflicting statutory or legal requirements or sufficient subscriptions to offset the cost.

<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>CRGR REVIEW</u>	<u>STATUS</u>
AEOD/PTB	J. Crooks	Low	N/A	Both publications are continuing. Termination is at present contingent on the consolidation efforts in (4) above, which is pending completion of the IE contract.

AEOD RECOMMENDATION TRACKING SYSTEM

RECOMMENDATION

SOURCE: Special Study Report AEOD/S603 (Memo dated June 6, 1986 from F. Hebdon to R. Vollmer)

Responsible AEOD Engineer: E. Leeds

TITLE OR SUBJECT: "Adequacy of the Scope of IE Bulletin 86-01"

RECOMMENDATION 1

IE should issue a supplemental or separate bulletin to address the potential single failure vulnerability of Westinghouse-designed PWR safety injection pumps.

	<u>RESPONSIBLE OFFICE/DIV/BR</u>	<u>CONTACT</u>	<u>PRIORITY</u>	<u>REVIEW</u>	<u>CRGR STATUS</u>
190	IE/DEPER/EAB	H. Bailey	N/A	10/86	Resolved. On October 8, 1986. IE issued IE Bulletin 86-03, "Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line" to address this recommendation.

APPENDIX A

SUMMARY OF 1986 ABNORMAL OCCURRENCES

(Including Those Approved by the
Commission and Those Still in the
Staff Office/Commission Approval Chain)

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
86-1	Loss of Power and Water Hammer Event	Vol. 9, No. 1	G-2	<p>On November 21, 1985, San Onofre Nuclear Generating Station (SONGS) Unit 1 experienced a partial loss of inplant ac electrical power while the plant was operating at 60 percent power. Following a manual reactor trip, the plant lost all inplant ac power for 4 minutes and experienced a severe water hammer in the feedwater system which caused a leak, damaged plant equipment, and challenged the integrity of the plant's heat sink. The most significant aspect of the event involved the failure of five safety-related check valves in the feedwater system, without detection, which jeopardized the integrity of safety systems. The event involved a number of equipment malfunctions, operator errors, and procedural deficiencies.</p> <p>The incident was investigated by an NRC Incident Investigation Team.</p>
86-2	Loss of Integrated Control System Power and Over- cooling Transient	Vol. 9, No. 1	G-3	<p>On December 26, 1985, Rancho Seco Nuclear Generating Station experienced a loss of dc power within the integrated control system (ICS) while the plant was operating at 76 percent power. Following the loss of ICS dc power, the reactor tripped on high reactor coolant system (RCS) pressure followed by a rapid overcooling transient and automatic initiation of the safety features actuation system on low RCS pressure. The overcooling transient continued until ICS dc power was restored 26 minutes after its loss. The significance of the event is that a nonsafety-</p>

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
A-3				related system failure initiated a plant transient which could have been more severe under other postulated scenarios.
				The incident was investigated by an NRC Incident Investigation Team.
	86-3 Rupture of a Uranium Hexafluoride Cylinder and Release of Gases	Vol. 9, No. 1	G and A-11	On January 4, 1986, a cylinder filled with uranium hexafluoride (UF-6) ruptured while it was being heated in a steam chest at the Sequoyah Fuels Corporation's Sequoyah Facility near Gore, Oklahoma. One worker died from pulmonary edema caused by inhalation of hydrofluoric acid, a reaction product of UF-6 and airborne moisture. Much of the facility complex and some offsite areas to the south were contaminated with hydrofluoric acid, and a second reaction product, uranyl fluoride. The interval of release was approximately 40 minutes.
				The incident was investigated by an NRC Augmented Investigation Team.
	86-4 Therapeutic Medical Misadministration	Vol. 9, No. 1	G	On February 7, 1986, a patient at Washington Hospital Center, Washington, D.C. received a cobalt-60 teletherapy treatment of 150 rads to the abdomen, which was intended for another patient.
	86-5 Overexposure to a Member of the Public from an Industrial Gauge	Vol. 9, No. 1	A-2	On February 19, 1986, while checking a licensee which had apparently ceased operations, an NRC Region III inspector determined that an industrial gauge, containing a sealed source of cobalt-60,

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
A-4	86-6 Breakdown of Management Controls at an Irradiator Facility	Vol. 9, No. 1	A-11	<p>was in an unrestricted area of the former factory site. Subsequent inspection determined that at least two members of the public received exposures to radiation as a result of the improper disposal of the gauge. The gauge had been licensed to C-E Glass, Inc., a Division of Combustion Engineering, Inc. The company operated a facility in St. Louis, Missouri, until October 1981.</p> <p>On March 3, 1986, the NRC issued an Order Suspending License (Effective Immediately) to Radiation Technology, Incorporated (RTI) of Rockaway, New Jersey. The Order was based on NRC inspections which identified a number of instances of bypassing safety interlock systems; these indicated a significant breakdown in the licensee's management control system.</p>
	86-7 Tritium Over-exposure and Laboratory Contamination	Vol. 9, No. 1	A-11	<p>During a routine inspection on March 12, 1986 at Ferris State College, Big Rapids, Michigan, an NRC inspector determined that, based on a review of bioassay test results, a licensee researcher had received an over-exposure of tritium (hydrogen 3) equivalent to a whole body exposure of about 21 rem, during an experiment on August 3, 1985. Continuing NRC inspections showed that two laboratories were contaminated. In addition, numerous deficiencies in the licensee's use and control of byproduct radioactive material were identified.</p>
AS86-1	Radiation Injury of an Industrial	Vol. 9, No. 1	A-1	<p>On April 20, 1984, an individual employed by BF Inspection Services in Midland, Texas, received an</p>

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
A-5	Radiographer (licensee of Texas, an Agreement State)			<p>exposure that resulted in a radiation burn while performing radiography in Seminole, Texas. The licensee failed to notify the Texas Bureau of Radiation Control (Agency) of the incident after it knew the radiographer had received a radiation burn. The burn was reported by Permian Industrial X-Ray, present employer of the individual, to the Agency on November 8, 1985, when the radiographer had an apparent recurrence of the wound.</p> <p>The radiographer may have received up to about 29,000 rems to his left hand and about 47 rems whole body exposure from the exposed source.</p>
	AS86-2 Contamination of a Scrap Steel Facility (licensee of California, an Agreement State)	Vol. 9, No. 1	G	<p>On May 24, 1985, it was discovered that some facilities of Tamco Steel Company of Ontario, California, were contaminated with radioactive material (later determined to be cesium). Apparently, the cesium (determined to be about 1.5 curies) was part of a radioactive device or source contained in a scrap steel shipment to the plant. The radioactivity was released when the scrap metal was melted in a furnace.</p>
	AS86-3 Radiation Injury of an Industrial Radiographer (licensee of Cali- fornia, an Agree- ment State)	Vol. 9, No. 1	A-1	<p>On August 25, 1985, an industrial radiographer received a radiation injury from an exposed source of his left hand (estimated to be about 2000 rads) and a whole body overexposure (estimated to be about 6 rads). At the time of the incident, the employee (employed by Boothe-Twining, Inc.)</p>

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
				was performing radiography at the company's field site in the Kern River oil field in Bakersfield, California. He was using a 46-curie iridium-192 source contained in a radiographic projector.
AS86-4	Radiation Injury of an Industrial Assistant Radiographer (licensee of Texas, an Agreement State)	Vol. 9, No. 1	A-1	<p>On November 9, 1985 an individual employed as an assistant radiographer by Basin Industrial X-Ray in Odessa, Texas, received a radiation burn of his left hand and an estimated 129 rems whole body exposure. The licensee failed to notify the Texas Bureau of Radiation Control (Agency) of the incident. Another licensee informed the Agency on November 26, 1985 that an incident had occurred involving Basin Industrial X-Ray.</p> <p>The estimated exposure to the employee's hand from the exposed source may have been as high as 30,000 rems, or even considerably higher.</p>
86-8	Out of Sequence Control Rod Withdrawal	Vol. 9, No. 2	A-11	On March 18, 1986, during a startup of Peach Bottom Unit 3, personnel errors by four licensed operators resulted in a control rod being withdrawn out-of-sequence without being detected by these operators. The next operating shift detected the error and manually scrammed the unit.
86-9	Boiling Water Reactor Emergency Core Cooling System	Vol. 9, No. 2	G-3, A-10, and	On May 19, 1986, the Boston Edison Company (BECO) notified the NRC that a significant design deficiency in the residual heat removal (RHR) system minimum

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
	Design Deficiency		A-12	<p>flow protection logic at the Pilgrim Nuclear Power Station (PNPS) had been discovered. Later, it was found that some other GE-designed BWRs (Dresden 2, 3 and Quad Cities 1, 2) also contained the same design deficiency.</p> <p>A similar deficiency was also discovered in some PWRs. This was discussed in an update to AO 86-9 in NUREG-0090, Vol. 9, No. 3.</p>
A-7	86-10 Willful Failure to Report a Diagnostic Medical Misadministration	Vol 9, No. 2	G and A-11	<p>On May 8, 1985, a patient at Mercy Hospital, Wilkes-Barre, Pennsylvania, received an injection of a radiopharmaceutical (a diagnostic dose of technetium-99m) intended for another patient. The misadministration was willfully not reported to the NRC as required by 10 CFR §35.43.</p>
	86-11 Therapeutic Medical Misadministration	Vol. 9, No. 2	G	<p>On April 9, 1986, at Maryview Hospital, Portsmouth, Virginia, a patient received a therapy dose in a chemical form other than that intended. This resulted in an unintended dose of several hundred rad to the patient's bone marrow. This could result in an increased chance of the patient contracting leukemia.</p>
	86-12 Willful Failure to Report Diagnostic Medical Misadministrations	Vol. 9, No. 2	G and 11	<p>On April 22, 1986, the NRC Office of Inspection and Enforcement issued an Order, effective immediately, removing a physician from the position of Radiation Safety Officer (RSO) and Authorized User at Bloomington Hospital, Bloomington, Indiana. The physician had willfully not reported five</p>

Abnormal Occurrences CY 1986

A-8

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
				diagnostic misadministrations; in addition, the physician obstructed the NRC inspection and misled the inspectors.
86-13	Diagnostic Medical Misadministration	Vol. 9, No. 2	G	On May 7, 1986, an outpatient of the Robert Packer Hospital and Guthrie Clinic in Sayre, Pennsylvania, received 10 millicuries of iodine-131 rather than the prescribed radio-pharmaceutical for a bone scan, technetium-99m. This would result in a considerable dose to the thyroid, which could result in reduced thyroid function.
86-14	Diagnostic Medical Misadministration	Vol. 9, No. 2	G	On June 17, 1986, at the Tripler Army Medical Center, Tripler AMC Hawaii, a patient received a dose of 3.09 mci of I-131 instead of a prescribed dose of 50 uci for a thyroid imaging procedure. The radiation exposure received by the patient due to the 3.09 mci I-131 dose is estimated to be 2472 rad to the thyroid, 0.43 rad to the ovaries, and 1.45 rad to the whole body. The dose to the thyroid could result in impairing its function.
AS86-5	Uncontrolled Release of Krypton-85 to an Unrestricted Area (licensee of Arizona, an Agreement State)	Vol. 9, No. 2	G-1 and A-3	On May 8, 1985, during routine operation of a Trio-Tech "Tracer-Flo" system at Micro-Rel Division, Medtronic, Incorporated, of Tempe, Arizona, a malfunction occurred which caused approximately 11.2 curies of radioactive krypton-85 to be vented into the atmosphere.

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
AS86-6	Contaminated Radio-pharmaceutical Used in Diagnostic Administrations (licensee of California, an Agreement State)	Vol. 9, No. 2	G	On May 9, 1985, a breakthrough of molybdenum-99 (a radioactive contaminant) occurred in a molybdenum-99/technetium-99m generator at Scripps Memorial Hospital, Encinitas, California. The breakthrough went unrecognized and the contaminated technetium-99m radiopharmaceutical was administered to four patients scheduled for diagnostic medical tests. Therefore, these patients received exposures higher than necessary.
(NUREG-0090, Vol. 9, No. 3 was sent to the Commission for approval on March 18, 1987, by SECY-87-73. It contains the following five proposed AOs and one proposed AO for NRC and Agreement State licensees, respectively.)				
86-15	Differential Pressure Switch Problem in Safety Systems at LaSalle Facility	Vol. 9, No. 3	G-2 and A-12	On June 1, 1986, LaSalle Unit 2 experienced a feed-water transient that resulted in low water level in the reactor vessel. The level reached a point where an automatic reactor scram would be expected; however, no such scram occurred. Subsequent investigation found that the problem was caused primarily by inadequate calibration of mechanical differential pressure switches supplied by SOR, Incorporated (formerly Static "O" Ring Pressure Switch Company). Similar switches have been installed in safety systems at many nuclear power plants.
86-16	Abnormal Cooldown and Depressurization Transient at Catawba Unit 2	Vol. 9, No. 3	B-4	On June 27, 1986, while Duke Power Company was conducting a startup test at Catawba Unit 2 from remotely located control panels, the reactor experienced an unexpected depressurization and

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
A-10				cooldown. There were no actual consequences to public health or safety. However, if the decay heat load of the reactor core had been greater and if the use of the remote shutdown panels had actually been required during a plant emergency, a more severe transient could have occurred.
	86-17 Significant Safe- guards Deficien- cies at Wolf Creek and Fort St. Vrain	Vol. 9, No. 3	A-8	On July 7, 1986, NRC Region IV issued enforcement letters containing Severity Level II violations to the licensees of two nuclear power stations for serious deficiencies in plant physical barriers. In the most serious example, it was determined at the Wolf Creek site that multiple uncontrolled access paths existed from the Owner Controlled Area (OCA) into the Protected Area (PA) and in two instances into Vital Areas (VAs). At the Fort St. Vrain site, NRC inspectors identified paths from the OCA to the PA and VA. In this situation, each access had a barrier installed, but each was evaluated to be inadequate and not capable of preventing an intruder from defeating it easily.
	86-18 Significant De- ficiencies in Access Controls at River Bend Station	Vol. 9, No. 3		On August 7, 1986, NRC Region IV issued an enforcement letter to the licensee containing a Severity Level II violation regarding serious deficiencies in controlling the access of personnel to vital areas. Conditions existed whereby an intruder could have obtained unauthorized and undetected access into vital areas from either the protected

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
86-19	Therapeutic Medical Vol. 9, No. 3 Misadministration		G	<p>area or other vital areas.</p> <p>During late August 1984, a high-activity iodine-125 radiation source, which had been implemented for treatment of a brain tumor in a patient of University of Cincinnati Medical Center, Cincinnati, Ohio, had leaked, causing an unintended radiation exposure of 2087 rad to the patient's thyroid. This dose could reduce the thyroid's function.</p> <p>Leakage occurred because one of the high activity seeds has been inadvertently cut into by a technician in the brachytherapy source storage room (BSR). The BSR was considerably contaminated and about 60 people received minor thyroid updates from airborne migration of the iodine-125.</p> <p>The event was not previously reported as an abnormal occurrence because at the time of the incident it was not classified as a medical misadministration as defined in 10 CFR §35.45. However, a reevaluation of the event by the NRC Staff during the latter part of 1986 concluded that the event should have properly been classified as a medical misadministration, and reportable as an abnormal occurrence.</p>

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
AS86-7	Therapeutic Medical Misadministration (licensee of Iowa, an Agreement State)	Vol. 9, No. 3	G	<p>On September 5, 1986, the Iowa Radiological Health Section, Bureau of Environmental Health, was notified of a therapeutic medical misadministration received by a patient at the University of Iowa Hospitals and Clinics, Iowa City, Iowa.</p> <p>The patient's bronchial tumor was being treated by an iridium-192 source placed in his bronchus tube. While sedated and asleep, the patient apparently pulled the tube containing the source out of the bronchus where it came to rest on his chest. The patient received an estimated 1500 rad to the chest in an area 3.4 cm long and 2 mm wide.</p>
(NUREG-0090, Vol. 9, No. 4 is under development by the NRC staff. The following nine events at NRC licensees are being considered by the staff for submittal to the Commission for approval.)				
86-20	Loss of Low Pressure Service Water Systems at Oconee Unit 2	Vol. 9, No. 4	G-3	<p>On October 1, 1986, while Unit 2 was in a refueling outage, the Unit 2 load shed test was twice performed. During both tests, the low pressure service water system pump suction was lost. Investigation showed that due to a design deficiency, the condenser circulating water system (which performs various safety-related functions) was degraded. A similar design deficiency existed on Units 1 and 3 which were operating at the time. These units were taken to cold shutdown until the problem was corrected.</p>

Abnormal Occurrences CY 1986

A-13

A0 #	Title of A0	Report No. NUREG-0090	A0 Criterion or Example	Comments
86-21	Degraded Safety Systems Due to Incorrect Switch Settings on Rotork Motor Operators at Catawba and McGuire Nuclear Stations	Vol. 9, No. 4	A-10	On October 23, 1986, the licensee discovered that many valves in safety systems were degraded at Catawba. On October 28, the licensee found a licensee using improper torque switch settings on the valve's Rotork motor operators. The problem was caused by the licensee using improper torque switch settings on the valve's Rotork motor operators. This could result in the valves not performing as designed (e.g., the activator motors switching off before the associated valves completed their travel).
86-22	Secondary System Pipe Break Resulting in Death of Four Persons at Surry Unit 2	Vol. 9, No. 4	A-10	On December 9, 1986, with both Units 1 and 2 at 100% power, Unit 2 tripped due to a low-low level in the "C" steam generator, followed by a rupture of an 18-inch suction line to the "A" train main feedwater pump. The reactor was taken to a cold shutdown condition with no release of radioactivity. However, eight individuals in the vicinity of the pipe rupture were injured due to the release of steam and water. Four of the individuals subsequently died. This incident was investigated by an NRC Augmented Investigation Team.
86-23	Release of Americium-241 Inside a Waste Storage Building at Wright-Patterson Air Force Base	Vol. 9, No. 4	G and A-11	On September 18 and October 6, 1986, a drum containing radioactive waste was opened to inspect its contents at Wright-Patterson Air Force Base, near Dayton, Ohio. Both openings resulted in a significant release of radioactive americium-

Abnormal Occurrences CY 1986

A-14

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
				241 inside the waste storage building, which significantly contaminated the building. The costs to date as a result of this incident is approximately \$500,000. Further costs will be incurred for either further decontamination of the building, or to dismantle and dispose of the building as radioactive waste.
86-24	Therapeutic Medical Misadministration	Vol. 9, No. 4	G	On October 6-8, 1986, a patient at the Cleveland Clinic Foundation, Cleveland, Ohio, received a series of therapeutic radiation exposures which resulted in a radiation dose of approximately 2000 rad (head to waist), instead of an intended dose of 1200 rad.
86-25	Suspension of License for Servicing Teletherapy and Radiography Units	Vol. 9, No. 4	A-11	On October 10, 1986, the NRC issued an Order suspending certain NRC-licensed service activities of Advanced Medical Systems, Inc., Geneva, Ohio. This action was taken after the NRC determined that the firm had been using untrained and unqualified employees to service cobalt-60 teletherapy units.
86-26	Diagnostic Medical Misadministration	Vol. 9, No. 4	G	On October 21, 1986, a patient at St. Luke's Hospital, Racine, Wisconsin, received a whole body iodine-131 diagnostic scan while the intended procedure was to be a thyroid scan. The whole body scan involved 1.53 millicuries of iodine-131, about 30 times the normal dosage for a thyroid scan. The patient may experience reduced thyroid function.

Abnormal Occurrences CY 1986

AO #	Title of AO	Report No. NUREG-0090	AO Criterion or Example	Comments
86-27	Diagnostic Medical Misadministration	Vol. 9, No. 4	G	On November 18, 1986, a patient at Toledo Hospital, Toledo, Ohio, received a misadministration of a radiopharmaceutical when the wrong radioactive material was administered. The patient's physician prescribed a bone scan, which normally involves about 20 millicuries of technetium-99m MDP. Instead, the patient received about 20 millicuries of iodine-131. As a result, the patient's thyroid received a dose of several thousand rad which is expected to significantly reduce the thyroid's function.
86-28	Immediately Effective Order Modifying License and Order to Show Cause Issued to an Industrial Radiography Company	Vol. 9, No. 4	G-3	On December 30, 1986, an Order was issued to Met-Chem Testing Laboratories of Utah, Inc., located in Salt Lake City, that removed a senior management employee from any assignment or position influencing or involving the performance or supervision of any NRC-licensed activities. This action was taken after the individual admitted he had typed a letter and forged on it the signature of a radiographer for the purpose of explaining away an overexposure indicated on the radiographer's film badge.

APPENDIX B
LISTINGS OF AEOD REPORTS
1980 - 1986

TABLE B-1
REPORTS ISSUED IN CY 1986

<u>Case and Special Studies</u>			
<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
8/86	Rupture of an Iodine-125 Brachytherapy Source at the University of Cincinnati Medical Center	C601	S. Pettijohn
8/86	Operational Experience Involving Turbine Overspeed Trips	C602	C. Hsu
12/86	A Review of Motor-Operated Valve Performance	C603	E. Brown
12/86	Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems	C604	M. Chirama1
12/86	Operational Experience Involving Losses of Electrical Inverters	C605	F. Ashe
1/86	Trends and Patterns Program Plan - FY86-FY88	P601	R. Dennig
8/86	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1985	P602	L. Bell
8/86	Trends and Patterns Report of Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Plants	P603	M. Harper
8/86	Trends and Patterns Analysis of the Operational Experience of Newly Licensed U.S. Nuclear Power Reactors	P604	T. Wolf
4/86	AEOD Annual Report for 1985	S601	J. Heltemes
5/86	An Overview of Nuclear Power Plant Operating Experience Feedback Programs	S602	J. Crooks
6/86	Adequacy of the Scope of IE Bulletin 86-01	S603	E. Leeds

Reactor Engineering Evaluations and Technical Reviews

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/9/86	Deficient Operator Actions Following Dual Function Valve Failures	E601	E. Leeds
1/15/86	Unexpected Criticality Due to Incorrect Calculation and Failure to Follow Procedures	E602	R. Freeman
2/19/86	Delayed Access to Safety-Related Areas During Plant Operation	E603	T. Cintula
3/14/86	Spurious System Isolations Caused by the Panalarm Model 86 Thermocouple Monitor	E604	E. Leeds
4/28/86	Lightning Events at Nuclear Power Plants	E605	M. Chiramal
5/27/86	Loss of Safety Injection Capability at Indian Point Unit 2	E606	R. Tripathi
5/27/86	Core Damage Precursor Event at Trojan (Revision 1)	E514	D. Zukor
7/3/86	Degradation or Loss of Charging Systems with Swing Pump Designs	E607	F. Ashe
7/14/86	Reexamination of Water Hammer Occurrences	E608	E. Leeds
8/8/86	Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation	E609	P. Lam
8/14/86	Loss of Low Pressure Coolant Injection Loop Selection Logic at Millstone Unit 1	E610	E. Leeds
10/16/86	Deficiencies in Seismic Anchorage for Electrical and Control Panels	E611	N. Thomasson
12/17/86	Emergency Diesel Generator Component Failures Due to Vibration	E612	C. Hsu
12/23/86	Localized Rod Cluster Control Assembly Wear at PWR Plants	E613	E. Brown

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/27/86	Pressure Sensitive Temperature Switch Results in Spurious Actuation of Fire Suppression System	T601	T. Cintula
4/29/86	Emergency Diesel Generator Cooling Water System Design Deficiencies at Maine Yankee and Haddam Neck	T602	E. Leeds
4/30/86	Inadvertent Pump Suction Transfer and Potential Auxiliary Feedwater Pump Cavitation at Davis-Besse	T603	R. Tripathi
5/7/86	Events Resulting from Deficiencies in Labeling and Identification Systems	T604	E. Trager
6/17/86	Failure of Main Steam Safety Valves to Properly Reseat	T605	R. Freeman
8/7/86	Inadvertent Recirculation Actuation Signals at Combustion Engineering Plants	T606	T. Cintula
9/19/86	Occurrence of Events Involving Wrong Units/Wrong Train/Wrong Component - Update through June 1986	T607	E. Trager
11/17/86	Hydrogen Fire and Failure of Detection System	T608	M. Chiramal
12/16/86	Foreign Material and Debris in Safety-Related Fluid Systems	T609	E. Leeds
12/19/86	ADS/RCIC System Interaction Events at River Bend Unit 1	T610	E. Leeds
12/19/86	Denied Access Due to Negative Room Pressure	T611	T. Cintula
12/31/86	Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches	T612	R. Tripathi

Nonreactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
6/17/86	Report on 1985 Nonreactor Events and Five-Year Assessment for 1981-1985	N601	K. Black
6/25/86	Medical Misadministrations Reported for 1985 and Five-Year Assessment of 1981-1985 Reports	N602	S. Pettijohn

Incident Investigation Program Reports

<u>Date</u>	<u>Subject</u>	<u>Designation</u>
1/86	Loss of Power and Water Hammer Event at San Onofre, Unit 1, on November 21, 1985	NUREG-1190
2/86	Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985	NUREG-1195
8/86	Incident Investigation Manual	--
12/86	Incident Investigation Manual - Revision 1	--

TABLE B-2
REPORTS ISSUED IN CY 1985

<u>Case and Special Studies</u>			
<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
6/85	Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	C501	H. Ornstein
9/85	Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors	C502	P. Lam
12/85	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	C503	H. Ornstein
12/85	Loss of Safety System Function Events	C504	E. Trager
12/85	Therapy Misadministrations Reported to NRC Pursuant to 10 CFR 35.42	C505	S. Pettijohn
6/85	Trends and Patterns Analysis of 1981 through 1983 LER Data (NUREG/CR-4129)	P502	B. Brady
7/85	Feedwater Transient Incidents in Westinghouse PWRs	P501	R. Dennig
8/85	Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Reactors January 1 through June 30, 1984	P503	T. Wolf
8/85	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1984	P504	L. Bell
3/85	Review of Operational Experience From Non-Power Reactors	S501	D. Zukor
4/85	AEOD Semiannual Report for July-December 1984	S502	J. Heltemes
9/85	Evaluation of Recent Valve Operator Motor Burnout Events	S503	E. Brown

Reactor Engineering Evaluations and Technical Reviews

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/17/85	Motor Operated Valve Failures Due to Hammering Problem	E501	M. Chiramal
1/25/85	Failure of Residual Heat Removal Suppression Pool Cooling Valve to Operate	E502	C. Hsu
3/4/85	Partial Failures of Control Rod Systems to Scram	E503	M. Chiramal
3/29/85	Loss or Actuation of Various Safety-Related Equipment Due to Removal of Fuses or Opening of Circuit Breakers	E504	F. Ashe
3/29/85	Service Water System Air Release Valve Failures	E505	S. Salah
5/13/85	Valve Stem Susceptibility to Intergranular Stress Corrosion Cracking Due to Improper Heat Treatment	E506	C. Hsu
5/17/85	Electrical Interaction Between Units During Loss of Offsite Power Event of August 21, 1984 at McGuire Units 1 and 2	E507	M. Chiramal
5/24/85	Nuclear Plant Operating Experience Involving Safety System Disturbances Caused By Bumped Electro-Mechanical Components	E508	S. Rubin
7/25/85	Salem Unit 2 Depressurization Event	E509	R. Freeman
7/30/85	Disabling of a Shared Diesel Generator Set Due to Electrical Power Supply Arrangement for Support Auxiliaries	E510	F. Ashe
8/9/85	Closure of Emergency Core Cooling System Minimum Flow Valves	E511	E. Leeds
9/4/85	Failure of Safety-Related Pumps Due to Debris	E512	R. Freeman
9/16/85	High Pressure Core Spray System Relief Valve Failures	E513	S. Salah
10/8/85	Core Damage Precursor Event at Trojan	E514	D. Zukor

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
12/11/85	Inadvertent Actuation of Safety System Due to Cross-Talk	E515	M. Chiramal
1/22/85	Failure of Automatic Protection for Boron Dilution Event at Callaway Unit 1	T501	R. Freeman
3/18/85	Comparative Analysis of Recent Feedline Water Hammer Events at Maine Yankee, Calvert Cliffs, Palisades, and Salem	T502	E. Leeds
5/2/85	Pressurizer Level Instrumentation of Combustion Engineering Reactor Units	T503	M. Chiramal
5/17/85	Loss of Instrument Air and Subsequent Pressure Transient at Callaway Unit 1	T504	R. Freeman
7/17/85	Beaver Valley Component Cooling Water Pump Damage	T505	C. Hsu
7/25/85	Primary System Release Due to Pressurizer Degas Relief Valve Lifting	T506	T. Cintula
8/13/85	Standby Liquid Control System Pressure Relief Valves Lift at a Pressure Lower than Reactor Coolant Pressure	T507	E. Brown
8/14/85	Browns Ferry Nuclear Plant High Pressure Coolant Injection System Performance Assessment	T508	E. Leeds
8/29/85	Inadequate Surveillance Testing Procedures for Degraded Voltage and Undervoltage Relays Associated with 4160-Volt Emergency Buses	T509	F. Ashe
9/4/85	Xenon Induced Power Oscillations at Catawba	T510	R. Freeman
9/16/85	Technicians Perform Work on Wrong Control Rod Drive Mechanism	T511	E. Trager
10/24/85	Incorrect Plugging of Steam Generator Tubes	T512	R. Freeman
11/7/85	Flooding of Safety-Related Valves in Pits	T513	D. Zukor

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
11/25/85	Potential Loss of Component Cooling Water Due to Maladjustment of Relief Valves	T514	D. Zukor
12/5/85	Residual Heat Removal Service Water Booster Pump Air Binding at Brunswick Unit 1	T515	S. Salah
12/11/85	High Pressure Coolant Injection Overspeed Trip Loss Events and Subsequent Damage Due to Water Hammer	T516	E. Trager

Nonreactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
5/20/85	Summary of the Nonreactor Event Report Data Base for the Period January-June 1984	N501	K. Black
6/28/85	Summary of the Nonreactor Event Report Data Base for the Period July-December 1984	N502	K. Black
7/25/85	Report on Medical Misadministrations for January 1984-December 1984	N503	S. Pettijohn

Incident Investigation Program Report

<u>Date</u>	<u>Subject</u>	<u>Designation</u>
7/85	Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985	NUREG-1154

TABLE B-3
REPORTS ISSUED IN CY 1984

<u>Case and Special Studies</u>			
<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
3/84	Low Temperature Overpressure Events at Turkey Point Unit 4	C401	W. Lanning
6/84	Operating Experience Related to Moisture Intrusion in Electrical Equipment at Commercial Power Reactors	C402	M. El-Zeftawy
5/84	Hatch Unit 2 Plant Systems Interaction Event on August 25, 1982	C403	S. Rubin
7/84	Steam Binding of Auxiliary Feedwater Pumps	C404	W. Lanning
9/84	Breaching of the Encapsulation of Sealed Well Logging Sources	C405	S. Pettijohn
2/84	Operating History Overview for Diesel Generators in Nuclear Service	P401	R. Dennig M. Chiramal
3/84	AEOD Trends and Patterns Program Plan	P402	R. Dennig
5/84	AEOD Trends and Patterns Evaluation Report, "Preliminary Assessment of LER Reporting Under 10 CFR 50.73"	P403	F. Hebdon
3/84	LER Data on Personnel Errors	P404	F. Hebdon
11/84	Draft Trends and Patterns Analysis of Feedwater Transients at Westinghouse PWRs	P405	M. Harper
11/84	Trends and Patterns Analysis of Reactor Scrams (Pilot Study)	P406	L. Bell
1/84	Human Error in Events Involving Wrong Unit or Wrong Train	S401	E. Trager

Case and Special Studies Continued

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
7/84	Pressure Locking of Flexible-Disk Wedge-Type Gate Valves	S402	S. Rubin
6/84	Annual Report of U.S. NRC Participation in the Nuclear Energy Agency Incident Reporting System During 1983	S403	J. Crooks
6/84	Analysis of Foreign IRS Reports Submitted During CY 1983	S404	D. Zukor
9/84	Semiannual Report on AEOD Activities	S405	J. Heltemes
10/84	Application of Risk Perspectives: A Procedures Guide	S406	P. Lam

Reactor Engineering Evaluations and Technical Reviews

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/4/84	Temporary Loss of All AC Power Due to Relay Failures in Diesel Generator Load Shedding Circuitry at Fort St. Vrain	E401	M. Chiramal
1/10/84	Water Hammer in Boiling Water Reactor High Pressure Coolant Injection Systems	E402	S. Rubin
1/17/84	Deficiency in Automatic Switch Company (ASCO) Spare Parts Kits for Scram Pilot Solenoid Valves	E403	F. Ashe
2/28/84	Failures in the Upper Head Injection System	E404	D. Zukor
3/22/84	Common Mode Failure of HPCI Steam Flow Isolation Capability at Browns Ferry	E405	M. El-Zeftawy
3/22/84	Mechanical Snubber Failure	E406	C. Hsu
3/26/84	Initiation and Indication Circuitry for High Pressure Coolant Injection Systems	E407	F. Ashe

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
3/27/84	Load Reduction Transient at the Salem Unit 2 on January 14, 1982 (Revision 1)	E323	N. Trehan
4/13/84	Reversed Differential Pressure Instrument Sensing Lines	E408	S. Rubin
5/16/84	Operating Experience Involving Air in Instrument Sensing Lines	E409	S. Salah
5/21/84	Operational Experiences Involving Standby Gas Treatment Systems which Illustrate Potential Common Cause Failure or Degradation Mechanisms	E410	F. Ashe
5/22/84	Failure of Anti-Cavitation Device in Residual Heat Removal Service Water Heat Exchanger Outlet Valve	E411	C. Hsu
5/25/84	Adverse System Interaction with Domestic Water Systems	E412	T. Cintula
5/25/84	Natural Circulation in Pressurized Water Reactors	E413	W. Lanning
5/31/84	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	E414	P. Lam
6/6/84	Overcooling Transient	E415	E. Imbro
6/11/84	Erosion in Nuclear Power Plants	E416	E. Brown
7/2/84	Loosening of Flange Bolts on Residual Heat Removal Heat Exchanger Leading to Primary to Secondary Side Leakage	E417	C. Hsu
7/24/84	Feedwater Transients During Startup at Westinghouse Plants	E418	D. Zukor
7/84	Failures of Fischer-Porter Transmitters Used in Safety-Related Systems	E419	M. Chiramal
8/23/84	Operational Experiences Involving Shorted Lamp Sockets of Indication Lights	E420	M. Chiramal

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
8/27/84	Loss of Pressurizer Heaters During Precore Hot Functional Testing	E421	T. Cintula
8/27/84	High Pressure Coolant Injection System Performance at Hatch Units 1 and 2	E422	T. Wolf
9/20/84	Failure of Large Hydraulic Snubbers to Lock Up	E423	E. Brown
10/1/84	Failure of Anchor Bolt on Diesel Generator Day Tank at Davis Besse Unit 1	E424	C. Hsu
10/11/84	High Pressure Coolant Injection System Lockout at Vermont Yankee	E425	M. Chiramal
10/24/84	Single Failure Vulnerability of Power Operated Relief Valve Actuation Circuitry for Low Temperature Overpressure Protection	E426	E. Imbro
11/6/84	Licensee Event Reports that Address Situations Which Potentially Could Result in Overloading Electrical Equipment in the Emergency Power System or Prevent Operation of the Onsite Power System Sequencer	E427	F. Ashe
3/2/84	Failures of Containment Air Monitors at Farley Units 1 and 2	T401	D. Zukor
3/22/84	Chemical Contamination of Primary and Secondary Systems in Light Water Reactors	T402	M. El-Zeftawy
3/23/84	Setpoint Drift of Barton Model 288 Switches	T403	M. Chiramal
4/13/84	Cable Fire and Loss of Control Power to Engineered Safeguards Valves	T404	M. Chiramal
4/25/84	Cold Weather Events 1983-1984	T405	T. Cintula
4/25/84	Improper Spare Parts Procurement Event at Grand Gulf Unit 1	T406	T. Wolf
4/30/84	Failure of 4 kV Circuit Breaker to Trip	T407	M. Chiramal

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
5/7/84	Diesel Generator Inoperability Due to Overheating of Ventilation Cowling	T408	M. Chiramal
5/1/84	Multiple Failures of Bell and Howell Dual Potentiometer Modules Which Occurred at the Fort Calhoun Nuclear Station	T409	F. Ashe
5/1/84	Injection Valve for the High Pressure Coolant Injection System Failure to Open During a Surveillance Test	T410	E. Brown
6/18/84	Contamination of the Nitrogen System at Sacramento Municipal Utility District	T411	M. El-Zeftawy
6/18/84	Failure of an Access Door Between the Drywell and the Wetwell	T412	T. Wolf
6/28/84	Failure of Fire Damper in Safeguards Ventilation System	T413	W. Lanning
7/12/84	Station Operating Restrictions for Lost or Out of Service Power Transformers through which Electrical Power is Supplied to the Emergency Buses	T414	F. Ashe
7/17/84	Destruction of Charging Pump	T415	W. Lanning
8/1/84	Loss of Engineered Safety Feature Auxiliary Feedwater Pump Capability at Trojan on January 22, 1983	T416	D. Zukor
8/2/84	Excessive Cooldown Rate Event at LaSalle Unit 1	T417	S. Salah
8/6/84	Events Involving Fires or Other Related Abnormalities in Motor Control Centers with Aluminum Bus Bars	T418	F. Ashe
8/20/84	Contamination of Snubber Bleed Screw and Lockup Poppet Valve	T419	C. Hsu
8/23/84	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure	T420	P. Lam

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
8/23/84	Design Deficiency in Standby Gas Treatment System	T421	M. Chiramal
8/29/84	Inoperability of Safety Injection Pump at Salem Unit 1 on October 17, 1983	T422	D. Zukor
10/25/84	Inoperability of Helium Circulator Over-speed Trip Channels Due to Impedance Variations in Speed Sensing Cables Exposed to Steam Leak	T423	E. Imbro
11/20/84	Fire Water Main Leakage Into 4 kV Switchgear Room at San Onofre Unit 1	T424	T. Cintula

Nonreactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
5/8/84	Report on Medical Misadministrations for January 1983 through June 1983	N204D	S. Pettijohn
6/11/84	Nonreactor Event Report Database for the Period July-December 1983	N401	K. Black
6/26/84	Events Involving Undetected Unavailability of the Turbine-Driven Auxiliary Feedwater Train	N402	E. Trager
7/3/84	Report on Medical Misadministrations for July 1983-December 1983	N403	S. Pettijohn

TABLE B-4
REPORTS ISSUED IN CY 1983

Case and Special Studies

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
4/83	Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	C301	M. Chiramal
9/83	Potentially Damaging Failure Modes of High and Medium Voltage Electrical Equipment	NUREG/CR-3122	M. Chiramal
7/83	Report on the Implications of the Anticipated Transient Without Scram Events at the Salem Nuclear Power Plant on the NRC Program for Collection and Analysis of Operational Experience	P301	J. Crooks

Reactor Engineering Evaluations and Technical Reviews

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/19/83	Fuel Degradation at Westinghouse Plants	E301	D. Zukor
1/31/83	Potential Loss of Service Water Flow Resulting from a Loss of Instrument Air	E302	E. Imbro
2/16/83	Valve Flooding Event at Surry	E303	D. Zukor
3/11/83	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments	E304	T. Cintula
4/13/83	Inoperable Motor Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	E305	E. Brown F. Ashe
4/14/83	Cooldown During Loss of Control Room Test at McGuire Unit 1	E306	D. Zukor
4/14/83	Update to AEOD/E301 (Fuel Degradation at Westinghouse Plants) (Revision 1)	E301	D. Zukor

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
4/18/83	Degradation of Safety-Related Batteries Due to Cracking of Battery Cell Cases and/or Other Possible Aging-Relating Mechanisms	E307	F. Ashe
4/19/83	Cracks and Leaks in Small Diameter Piping	E308	E. Brown
4/21/83	The Potential for Water Hammer During the Restart of Residual Heat Removal Pumps at BWR Nuclear Power Plants	E309	S. Rubin
4/25/83	Loss of Shutdown Cooling and Subsequent Boron Dilution at San Onofre Unit 2	E310	T. Cintula
4/25/83	Loss of Salt Water Flow to the Service Water Heat Exchangers for 23 Minutes at Calvert Cliffs Unit 2	E311	T. Cintula
5/18/83	Operability of Target Rock Safety Relief Valves in the Safety Mode with Pilot Valve Leakage	E312	J. Pellet
6/15/83	Potential Contamination of the Spent Fuel Pool and Primary Reactor System	E313	E. Brown
6/28/83	Loss of All Three Charging Pumps Due to Empty Common Reference Leg in the Liquid Level Transducers for the Volume Control Tank at St. Lucie 1	E314	T. Cintula
7/5/83	Misuse of Valve Resulting in Vibration and Damage to the Valve Assembly and Pipe Supports	E315	E. Brown
7/11/83	Frozen Ice Condenser Intermediate Deck Doors	E316	D. Zukor
8/1/83	Loss of High Pressure Injection System	E317	N. Trehan
8/15/83	Biofouling at Salem Units 1 and 2	E318	E. Imbro
9/8/83	Loss of Drywell-Torus Pressure Differential During Residual Heat Removal Pump Flow Testing at Cooper	E319	S. Rubin
9/8/83	Power Operated Relief Valve (PORV) Actuation Resulting in Safety Injection Actuation at Calvert Cliffs	E320	E. Imbro

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
9/12/83	Three Similar Events of a Loss of Shutdown Cooling Flow at Combustion Engineering Plants	E321	T. Cintula
9/16/83	Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting at Peach Bottom Unit 2	E322	C. Hsu
9/19/83	Load Reduction Transient at Salem Unit 2 on January 14, 1982	E323	N. Trehan
9/21/83	Review of Events Involving Failures of Power Supply in Instrumentation and Control Systems	E324	M. Chiramal
11/21/83	Vapor Binding of Auxiliary Feedwater Pumps at Robinson Unit 2	E325	W. Lanning
11/28/83	Steam Voiding in Oconee Unit 3 on June 13, 1975: A Precursor Event to the TMI-2 Accident	E326	H. Ornstein
11/28/83	Gaseous Releases From Waste Gas Disposal System	E327	N. Trehan
1/19/83	Diesel Generator Load Sequencer Design Deficiency - LER 82-025/OIT	T301	M. Chiramal
2/9/83	Postulated Loss of Auxiliary Feedwater System Resulting from a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	T302	E. Imbro
3/2	Seat Degradation in Henry Pratt Butterfly Valves	T303	E. Brown
3/23	Cause of Containment Isolation Valve F042A to Close at Brunswick Unit 1	T304	S. Salah
3/28	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Intrusion at Sequoyah Unit 1	T305	E. Imbro

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
4/13/83	Scram Discharge Volume Level Switch Failure at Hatch Unit 2	T306	J. Pellet
4/19/83	Condensate Demineralizer Resin Migration through the Plant Vent and the Standby Gas Treatment System at Pilgrim Unit 1	T307	J. Pellet
4/20/83	Undetectable Failure in Westinghouse Solid State Protection System	T308	M. Chiramal
4/25/83	Air in Reactor Water Cleanup System Instrument Sensing Lines at Brunswick Unit 2	T309	S. Salah
4/25/83	Blocking of Automatic Safety Injection Signals	T310	M. Chiramal
5/5/83	Rod Control Urgent Failure on June 25, 1982 at Surry Unit 2	T311	N. Trehan
5/9/83	Failure of 5 kV Cable Terminations	T312	M. Chiramal
5/11/83	Capped Containment Pressure Sensing Lines	T313	S. Rubin
5/24/83	Improper Size of Inlet Piping to Primary Safety Valves	T314	E. Imbro
5/24/83	Events Involving Losses of or Perturbations in a Single 120 Volt AC Vital Power Supply Inverter and Attendant Distribution Bus which Resulted in Inadvertent Actuations of Safety Systems	T315	F. Ashe
5/31/83	Thermal Nonrepeatability Problem with Barton Model 763 and 764 Electronic Transmitters	T316	M. Chiramal
6/13/83	Problems with Diesel-Driven Containment Spray Pump at Zion Unit 2 on December 16, 1982	T317	D. Zukor
6/13/83	Failure of Recirculation Spray Service Water Motor Operated Valves	T318	D. Zukor

Reactor Case Studies and Engineering Evaluations (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
6/13/83	Design Deficiency in Control Circuits of Feedwater Isolation Valves and Boron Injection Tank Recirculation Valves	T319	M. Chirama1
6/14/83	Inadvertent Safety Injections Attributed to Personnel Error at Summer	T320	F. Ashe
6/15/83	Check Valve Installed Backwards in Instrument Air Line to the Power Operated Relief Valve at Surry Unit 1	T321	D. Zukor
6/15/83	Gouges in Main Coolant System Piping at Diablo Canyon on April 19, 1983	T322	D. Zukor
6/17/83	Turbine Trip Bypass Delay at Grand Gulf Unit 1	T323	S. Salah
7/27/83	Events Involving Two or More Simultaneously Dropped Rod Control Cluster Assemblies	T324	F. Ashe
8/1/83	Leakage in Static-O-Ring Pressure Switches	T325	M. Chirama1
8/2/83	Safety/Relief Valve Corrosion at a Foreign Reactor	T326	E. Brown
8/11/83	Auxiliary Feedwater Header Problems at Babcock & Wilcox Plants	T327	H. Ornstein
8/12/83	Two of Three Emergency Core Cooling System Accumulators Inoperable at Surry Unit 1	T328	D. Zukor
8/24/83	Leak in Reactor Water Cleanup System "B" Regenerative Heat Exchanger Relief Line	T329	C. Hsu
8/29/83	Steam Generator Tube Rupture at Oconee Unit 2	T330	M. El-Zeftawy
8/29/83	Review of Events at Operating Nuclear Plants Involving Plant Computers	T331	M. Chirama1
10/7/83	Reactor Vessel Drainage at Grand Gulf Unit 1	T332	S. Salah

Reactor Engineering Evaluations and Technical Reviews (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
10/31/83	Degradation of Saltwater Cooling System at San Onofre Unit 1 Caused by a Loss of Instrument Air	T333	H. Ornstein
11/15/83	Reactor Vessel Drainage at Grand Gulf Unit 1	T334	S. Salah
11/15/83	Simultaneous Safety Injection Actuation Signal and Recirculation Actuation Signal at San Onofre Unit 3	T335	E. Imbro
11/17/83	Design Deficiency Resulting in Isolation of Both Loops of the Emergency Condenser System at Nine Mile Point Unit 1	T336	M. Chirama1
11/21/83	Water Hammer in the Main Feedwater System Resulting in a Feedwater Line Crack at Maine Yankee	T337	E. Imbro
11/28/83	Water Leak through Containment Spray Block Valves at San Onofre 1	T338	D. Zukor
11/29/83	Redundant Emergency Core Cooling System Pump Room Air Coolers Out of Service for 22 Hours at Calvert Cliffs Unit 1	T339	T. Cintula
12/2/83	Evaluation of a Control Rod Mismanipulation Event at Hatch Unit 2	T340	T. Wolf
12/19/83	Corrosion of Carbon Steel Pipe in Service Water Headers	T341	E. Brown

Nonreactor Engineering Evaluations and Technical Reviews

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/11/83	Nonreactor Event Report Database for the Period January-June 1982	N209A	E. Trager
3/18/83	I125/I131 Effluent Releases by Material Licensees	N301	S. Pettijohn

Nonreactor Engineering Evaluations and Technical Reviews

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
6/10/83	Mound Laboratory Fabricated PuBe Sources	N302	K. Black
6/10/83	Americium Contamination Resulting from Rupture of Well-Logging Sources	N303	K. Black
6/14/83	Nonreactor Event Report Database for the Period July-December 1982	N209B	K. Black
7/14/83	Americium-241 Sources	N304	
7/14/83	Report on Medical Misadministrations for January 1981-December 1982	N204C	S. Pettijohn
8/4/83	Human Factors Contributions to Accident Sequence Precursor Events	N305	E. Trager
12/1/83	Potentially Leaking Americium-241 Sources Manufactured by Amersham Corporation	N306	S. Pettijohn
12/28/83	Nonreactor Event Report Database for the Period January-June 1983	N307	K. Black
3/10/83	Internal Exposure to Am-241	NT301	K. Black
4/5/83	Kay-Ray, Inc. Reports of Suspected Leaking Sealed Sources Manufactured by General Radioisotope Products	NT302	S. Pettijohn
8/24/83	Possession of Unauthorized Sealed Source/ Exposure Device Combinations by Mid-Con Inspection Services, Inc.	NT303	S. Pettijohn
11/4/83	Human Factors Involvement in Events at Oconee Units 1, 2, and 3	NT304	K. Black

TABLE B-5
REPORTS ISSUED IN CY 1982

Case Studies

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/82	Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors	C201	M. Chiramal
2/82	Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick	C202	E. Imbro
5/82	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980	C203	E. Brown
7/82	San Onofre Unit 1 Loss of Salt Water Cooling Event on March 10, 1980	C204	H. Ornstein
8/82	Abnormal Transient Operating Guidelines as Applied to the April 1981 Overfill Event at Arkansas Nuclear One, Unit 1	C205	J. Pellet
10/82	Inadvertent Loss of Reactor Coolant Events at the Sequoyah Nuclear Plant, Units 1 and 2	C206	W. Lanning

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/12/82	Methodology for Vital Area Determination	E201	W. Lanning
1/13/82	Loss of High Pressure Injection Lube Oil Cooling at Rancho Seco	E202	J. Pellet
1/21/82	Inadvertent Isolation of Containment Fan Units at Salem Unit 1	E203	W. Lanning
1/28/82	Effects of Fire Protection System Actuation on Safety-Related Equipment	E204	M. Chiramal

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
2/16/82	Potential Consequences of Heavy Load Drop Accidents in LWRs	E205	M. El-Zeftawy
2/22/82	Load Reduction Transient on January 14, 1982 at Salem Unit 2	E206	N. Trehan
2/22/82	LER 50-336/81-26: Investigation of the Relative Frequency of Valve Overtravel Anomalies that Could Result in a Potential Centrifugal Pump Runout Exceeding Net Positive Suction Head	E207	E. Imbro
2/22/82	An Observed Difference in Lift Setpoint for Steam Generator and Pressurizer Safety Valves	E208	E. Imbro
2/23/82	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback Unit 1 on April 13, 1979)	E209	M. Chiramal
2/23/82	Inadequate Switchgear Cooling at Beaver Valley Unit 1	E210	W. Lanning
2/24/82	Repetitive Failures of Emergency Feedwater Flow Valves at Arkansas Unit 2 Because of Valve Operator Hydraulic Problems	E211	E. Imbro
2/24/82	Spurious Trip of the Generator Lockout Relay Associated with a Diesel Generator Unit	E212	F. Ashe
2/24/82	Trip of Two Inservice Auxiliary Feedwater Pumps from Low Suction at Zion Unit 2 on December 11, 1981	E213	D. Zukor
3/1/82	Duane Arnold Loss of River Water System Loop	E214	T. Wolf
3/18/82	Engineering Evaluation of the Salt Service Water System Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels	E215	E. Imbro
3/28/82	A Recently Evaluated Preoperational Test Precursor of the TMI-2 Accident	E216	H. Ornstein

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
3/31/82	Scram Pilot Solenoid Valve Failures Due to Low Voltage - Grand Gulf Unit 1	E217	M. Chiramal
3/31/82	Potential for Air Binding or Degraded Performance of BWR Residual Heat Removal System Pumps During the Recirculation Phase of a Loss-of-Coolant Accident	E218	S. Rubin
4/1/82	Proposed Circular: Contamination of Air Serving Safety-Related Equipment	E219	H. Ornstein
4/6/82	Water in the Fuel Oil Tank at Surry Power Station Unit 2	E220	N. Trehan
4/22/82	Indian Point Unit 2 Flooding Event	E221	W. Lanning
5/10/82	Loss of Reserve Station Service Transformer "B" on January 18, 1982 at Surry Unit 2	E222	N. Trehan
5/11/82	Inadvertent Loss of Coolant Events at Sequoyah Units 1 and 2	E223	W. Lanning
5/21/82	Generic Concerns Associated with the Ginna Steam Generator Tube Rupture Event	E224	W. Lanning
6/1/82	Degradation of BWR Scram Pilot Solenoid Valves Due to Abnormal Power Supply Voltage	E225	M. Chiramal
6/18/82	Inoperability of Instrumentation Due to Extreme Cold Weather	E226	M. Chiramal
6/24/82	Failure of Engineered Safety Features Manual Initiation Pushbutton Switches	E227	F. Ashe
6/25/82	Repetitive Overspeed Trips of the Steam Driven Emergency Feedwater Pump on Initial Start at Arkansas Nuclear One, Unit 2	E228	E. Imbro
6/29/82	Potential for Flooding in Control Room at San Onofre Units 2 and 3	E229	E. Imbro
7/7/82	Water in the Fuel Oil Tank at Surry Power Station, Unit 2 - Additional Information	E230	N. Trehan

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
7/19/82	Millstone Unit 2 Loss of Shutdown Cooling Due to Trip of Low Pressure Safety Injection Pump	E231	M. Chiramal
7/19/82	Potential Deficiency in the Sigma Lumigraph Indicators Model Number 9270	E232	F. Ashe
7/28/82	Carbon Dioxide Systems Used for Fire Protection in or Adjacent to Critical Areas	E233	M. Chiramal
8/11/82	Failure in a Section of 4 kV Bus Cable Manufactured by Okonite	E234	F. Ashe
8/11/82	Wiring Error in Handswitch for Solenoid Control Valves Associated with High Pressure Coolant Injection System Steam Condensing Mode Pressure Control Valve at Duane Arnold	E235	S. Rubin
8/25/82	Brunswick Steam Electric Plant Unit 2 Loss of Residual Heat Removal Service Water on January 16, 1982	E236	T. Wolf
8/25/82	Power Operated Relief Valve Failure at Robinson	E237	E. Brown
8/25/82	Water in the Lube Oil in Safety Injection Pump IA-A at Sequoyah - LER 81-076	E238	N. Trehan
9/24/82	Main Steam Isolation Valve Closures and Pressurizer Safety Valve Actuations at St. Lucie Unit 1 on December 19, 1981	E239	T. Cintula
9/29/82	Preliminary Account of Events Associated with a Reactor Trip at Hatch Unit 2 on August 25, 1982	E240	S. Rubin
10/1/82	Emergency Diesel Generator System Problems at Fitzpatrick	E241	M. Chiramal
10/21/82	Fuel Assembly Degradation while in the Spent Fuel Storage Pool	E242	E. Brown

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
10/21/82	Plant Trip Followed by a Safety Injection Caused by Loss of "A" Cooling Tower Pump at Palisades on February 4, 1982	E243	T. Cintula
10/21/82	Loss of Residual Heat Removal System Event at Pilgrim Nuclear Power Station on December 21, 1981	E244	T. Wolf
10/21/82	Failure of Westinghouse Type SC-1 No. 1876-072 Relays	E245	F. Ashe
10/21/82	Events Involving Loss of Electrical Inverters Including Attendant Inverters to Vital Instrument Buses	E246	F. Ashe
10/26/82	Engineering Evaluation of Turbine/Reactor Trip at Rancho Seco on August 7, 1981	E247	J. Pellet
11/2/82	Engineering Evaluation Report on McGuire Overpressurization Event of August 27, 1981	E248	D. Zukor
11/4/82	Engineering Evaluation Memorandum, Licensee Reporting of the Turbine/Reactor Trip at Rancho Seco on August 7, 1981	E249	H. Ornstein
11/8/82	Quad Cities Unit 2 Loss of Auxiliary Electrical Power Event on June 22, 1982	E250	M. Chiramal
11/9/82	Salem Unit 2 Loss of Vital Bus No. 2A	E251	M. Chiramal
11/17/82	Potential Control Logic Problem Resulting in Inoperable Auto-Start of Diesel Generator Units Under the Conditions of Loss-of-Coolant Accident and Loss of Station	E253	F. Ashe
11/17/82	Review of Prairie Island Unit 1 LER 82-015/01T on Diesel Generator Operability	E254	M. Chiramal

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
11/17/82	Failure of the Vent Line on the Common Discharge of the Two Motor-Driven Auxiliary Feedwater Pumps at San Onofre Unit 2 from an Improper Valve Lineup	E255	T. Cintula
11/24/82	Loss of Shutdown Cooling and Subsequent Boron Dilution at San Onofre Unit 2	E256	T. Cintula
12/2/82	Insufficient Net Positive Suction Head for Charging Pump Service Water Pumps at Surry Nuclear Power Station	E257	D. Zukor

Nonreactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
2/1/82	Report on Medical Misadministrations for the Period November 10, 1980 - September 30, 1981	N201	S. Pettijohn
1/21/82	Buildup of Uranium-Bearing Sludge in Waste Retention Tanks	N202	K. Black
2/18/82	Lost Plutonium - 238 Source	N203	K. Black
3/5/82	Report on Medical Misadministrations for CY 1981	N204	S. Pettijohn
4/27/82	Preliminary AEOD Review of Iodine-125 Sealed Source Leakage Incidents	N205	E. Trager
5/6/82	Eberline Instrument Corporation - Part 21 Report	N206	K. Black
5/25/82	AEOD Review of Iodine-125 Sealed Source Leakage Incidents	N207	E. Trager
8/2/82	Potentially Leaking Plutonium-Beryllium Neutron Sources	N208	S. Pettijohn

Nonreactor Engineering Evaluations (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
8/30/82	A Summary of the Nonreactor Event Report Database for 1981	N209	K. Black
11/15/82	Leaking Hoses on Self Contained Breathing Apparatus (SCBA) Manufactured by MSA	N210	K. Black

TABLE B-6
REPORTS ISSUED IN CY 1981

<u>Case Studies</u>			
<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
3/81	Report on the St. Lucie Unit 1 Natural Circulation Cooldown on June 11, 1980	C101	E. Imbro
3/81	Robinson Reactor Coolant System Leak on January 29, 1981	C102	W. Lanning
3/81	AEOD Safety Concerns Associated with Pipe Breaks in the BWR Scram System	C103	S. Rubin
4/81	Millstone Unit 2 Loss of 125 V DC Bus Event on January 2, 1981	C104	M. Chiramal
12/81	Report on the Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980	C105	E. Imbro

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
1/19/81	Degradation of Internal Appurtenances in LWR Piping	E101	E. Brown
1/30/81	Sequoyah Unit 1 Loss of Annunciation	E102	M. Chiramal
3/2/81	Engineering Evaluation of Feedwater Transient and System Pipe Break at Turkey Point 3	E104	S. Sands
3/31/81	Water Hammer During Restart of Residual Heat Removal Pumps	E105	J. Huang
3/31/81	Water Hammer in the Steam Condensing Mode of the Residual Heat Removal System Operation	E106	J. Huang
4/17/81	Peach Bottom Unit 3 Occurrence on February 25, 1981	E107	F. Ashe

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
4/21/81	Hatch Units 1 and 2 - Alternate Offsite Source Interlock with Emergency Diesel Generators	E108	M. Chiramal
4/24/81	Potential Common Mode Failure of Diesel Generators	E109	M. Chiramal
4/29/81	Requirements of the Preferred or Offsite Power System	E110	F. Ashe
5/22/81	Evaluation of High Pressure Safety Injection Pump Operability without Service Water	E111	E. Imbro
6/15/81	Inoperability of Instrumentation Due to Extreme Cold Weather	E112	M. Chiramal
6/24/81	Deliberate Pump Trip at Browns Ferry Unit 2 on April 6, 1981	E113	W. Lanning
6/24/81	Control System Failures that Could Cause or Exacerbate Nuclear Power Plant Accidents	E114	F. Ashe
7/8/81	Additional Information on Events at TMI-2 During Preoperational Testing (9/5-12, 1977)	E115	H. Ornstein
7/14/81	Failure of B Phase Main Transformer and Subsequent Fire in the Transformer Area - North Anna Unit 2	E116	M. Chiramal
7/16/81	Events at TMI-2 During Preoperational Testing	E117	H. Ornstein
7/20/81	Setpoint Drift Occurrences for the Barton Model 28B Instrument	E118	F. Ashe
7/22/81	Loss of Residual Heat Removal Capability at Brunswick Units 1 and 2	E119	E. Imbro
8/6/81	Ignition of Gaseous Waste Decay Tank at San Onofre Unit 1 - July 17, 1981	E120	H. Ornstein
8/28/81	Crystal River 3 Engineered Safeguards Relay Failures	E121	M. Chiramal

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
9/4/81	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of Integrated Control System Nonnuclear/Instrumentation	E122	H. Ornstein
9/15/81	Immediate Action Memo: Common Cause Failure Potential at Rancho Seco - Desiccant Contamination of Air Lines	E123	H. Ornstein
9/24/81	Review of Information on Purge Valves	E124	E. Brown
10/15/81	Engineering Evaluation Report on Shutdown Cooling System Heat Exchanger Failures at Oyster Creek, August 1981	E125	G. Lanik
10/16/81	Event Sequences Not Considered in the Design of Emergency Bus Control Logic	E126	F. Ashe
10/28/81	Pressure Boundary Degradation Due to Pump Seal Failure at Arkansas Nuclear One	E127	W. Lanning
11/10/81	Inoperable Teledyne Solenoid Valves	E128	F. Ashe
12/7/81	Brunswick Unit 2 Diesel Generator Jacket Water Temperature Control Valve and Manual Bypass Valve	E129	M. Chiramal
12/7/81	Davis Besse LER 79-062 on Auxiliary Feedwater System Pressure Switches	E130	M. Chiramal
12/10/81	High Circulating Current Associated with Inverter Output Due to Lack of Circuit Tuning	E131	F. Ashe
12/23/81	Abnormal Wear Encountered on Aloyco Swing Check Valves Installed in the Low Pressure Safety Injection System at Palisades	E132	T. Cintula
4/15/81	Inadequacies in Periodic Testing of Combustion Engineering PWR Reactor Protection System	E133	M. Chiramal

Nonreactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
3/16/81	Interim Report on Brown Boveri Betatron Calibration Check Source	N101	E. Trager
3/26/81	Irradiator Incident at an Agreement State Licensee's Facility (Becton-Dickinson, Broken Bow, Nebraska)	N102	K. Black
4/13/81	Interim Report on the October 1980 Fire at the Sweetwater Uranium Mill	N103	E. Trager
4/30/81	Interim Report on the January 2, 1981 Fire at the Atlas Uranium Mill	N104	E. Trager
5/18/81	Interim Report on Tailings Impoundment Liner Failure at the Sweetwater Uranium Mill	N105	E. Trager
8/12/81	Review of Reports of Leaking Radioactive Sources	N106	E. Trager
12/10/81	Engineering Evaluation of Fire Protection at Nonreactor Facilities	N107	E. Trager
12/16/81	Notes on AEOD Review of Emissions from Tritium Manufacturing and Distribution Licensees	N108	E. Trager

TABLE B-7
REPORTS ISSUED IN CY 1980

<u>Case Studies</u>			
<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
7/80	Report on the Browns Ferry Unit 3 Partial Failure to Scram Event on June 28, 1980	C001	S. Rubin
9/80	Report on the Interim Equipment and Procedures at Browns Ferry Unit 3 to Detect Water in the Scram Discharge Volume	C002	G. Lanik
10/80	Report on Loss of Offsite Power Event at Arkansas Nuclear One, Units 1 and 2	C003	W. Lanning
11/80	AEOD Actions Concerning the Crystal River Unit 3 Loss of Nonnuclear Instrumentation and Integrated Control System Power on February 26, 1980	C004	H. Ornstein
12/80	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	C005	E. Imbro
<u>Reactor Engineering Evaluations</u>			
<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
3/20/80	Crystal River Nuclear Power Plant Decay Heat Closed Cycle Cooling Water Pumps/DCP-IA and DCP-IB	E001	H. Ornstein
5/23/80	BWR Jet Pump Integrity	E002	S. Rubin
6/19/80	Comparison of Reactor Coolant Pump Events Contained in LERs, NPRDS, RECON, and Plant Records	E003	E. Brown
7/10/80	Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants, January 1, 1972 to April 30, 1978	E004	H. Ornstein

Reactor Engineering Evaluations (Continued)

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
7/15/80	Operational Restrictions for Class 1E 120 V AC Vital Instrument Buses	E005	M. Chirama1
8/4/80	Loss of Residual Heat Removal at Beaver Valley, LER 80-031	E006	W. Lanning
8/18/80	Potential for Unacceptable Interaction Between the Control Rod Drive System and Nonessential Control Air System at Browns Ferry	E007	S. Rubin
8/20/80	Operational Restrictions During Surveillance Testing of Emergency Diesel Generators	E008	M. Chirama1
8/22/80	Failure of Containment Isolation Valves at Zion	E009	W. Lanning
8/27/80	Tie Breaker Between Redundant Class 1E Buses - Point Beach Units 1 and 2	E010	M. Chirama1
8/29/80	Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment	E011	E. Imbro
9/12/80	Salem Unit 1 - Solenoid Valve of Containment Fan Coil Unit Service Water Flow Control Valve	E012	M. Chirama1
9/12/80	Excessive Main Feedwater Transient	E013	J. Creswell
10/8/80	Transient at Crystal River Unit 3 - September 30, 1980	E014	H. Ornstein
10/20/80	January 3, 1977 Quad Cities Unit 1 Loss of Air Event and its Effects on Scram Capability	E015	G. Lanik
10/21/80	Flow Blockage in Essential Equipment at ANO Caused by <u>Corbicula</u> sp. (Asiatic Clams)	E016	E. Imbro
10/29/80	Engineering Evaluation of Steam Generator Overfill	E017	W. Lanning

Reactor Engineering Evaluations

<u>Date</u>	<u>Subject</u>	<u>No.</u>	<u>Author</u>
12/18/80	Potential Failure of BWR Backup Scram (Mode Switch in Shutdown) Capability	E018	M. Chiramal
12/19/80	Davis Besse Unit 1 - Emergency Core Cooling System Actuation During Hot Shutdown on December 5, 1980	E019	M. Chiramal
12/24/80	Internal Appurtenances in LWRs	E020	E. Brown